96

44

547

ORNL-3969
UC-41 – Health and Safety

HEALTH PHYSICS AND SAFETY
ANNUAL REPORT FOR 1965

OAK RIDGE NATIONAL LABORATORY · U.S. ATOMIC ENERGY COMMISSION

OAK RIDGE NATIONAL LABORATORY
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

Printed in USA. Price $3.00 . Available from the Clearinghouse for Federal Scientific and Technical Information, National Bureau of Standards, U.S. Department of Commerce, Springfield, Virginia 22151

ORNL-3969

Contract No. W-7405-eng-26

HEALTH PHYSICS DIVISION

HEALTH PHYSICS AND SAFETY ANNUAL REPORT FOR 1965

K. Z. Morgan, Director

W. S. Snyder, Assistant Director

D. M. Davis, Section Chief

JULY 1966

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U. S. ATOMIC ENERGY COMMISSION

TABLE OF CONTENTS

TABLE OF CONTENTS (continued)

1.0 LIST OF TABLES

2.0 LIST OF FIGURES

2.0 LIST OF FIGURES (continued)

3.0 ACKNOWLEDGMENTS

The data for this report were contributed by: H. H. Abee, Section Chief of the Environs Radiation Monitoring Section; R. L. Clark, Section Chief of the Health Physics and Safety Surveys Section; E. D. Gupton, Section Chief of Applied Radiation Dosimetry Section; and A. D. Warden, Associate Department Head of Health Physics and Safety. Section 9.5, Whole Body Counter, was contributed by B. R. Fish, Section Chief of the Health Physics Technology Section.

4.0 SUMMARY

In August, 1965 the Applied Health Physics Section was delegated responsibilities for industrial safety which are similar to their responsibilities for radiation safety. Subsequent to this increase in responsibilities, the name of the section was changed from Applied Health Physics to Health Physics and Safety.

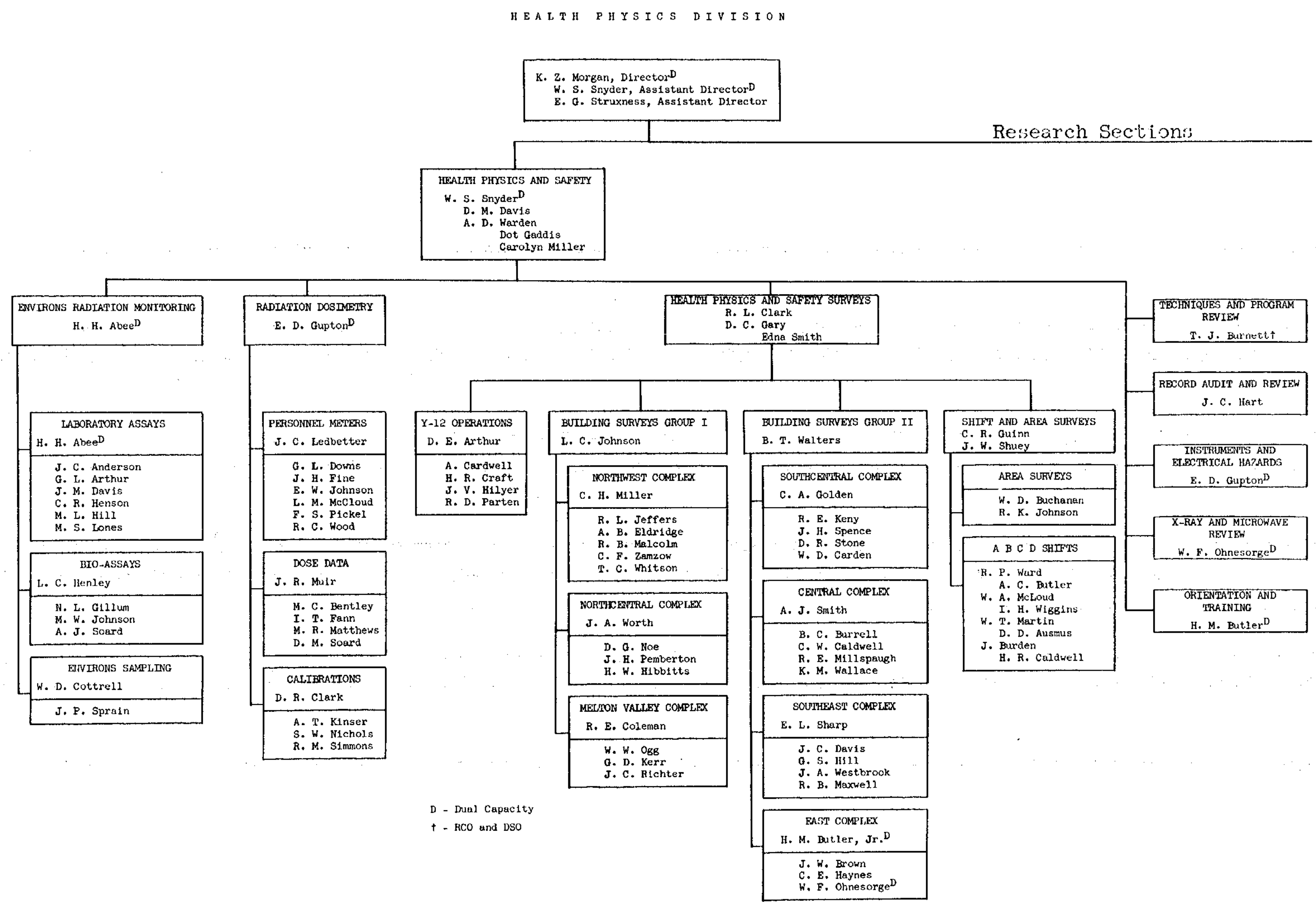
The gaseous and liquid waste releases from the Laboratory were such that the concentration of radioactive materials in the environs was well below the maximum levels recommended by the NCRP and FRC. The average concentration of radioactive materials in the atmosphere at the X-10 site was less than one percent of the maximum permissible for persons residing in the neighborhood of an atomic energy installation, and the concentration was, as expected, even less at the perimeter of the controlled area. The calculated average concentration of radioactive materials in the Clinch River at the point of entry of White Oak Creek into the River was also less than one percent of the maximum permissible for persons residing in the neighborhood of an atomic energy installation.

No employee received an external or internal radiation dose which exceeded the maximum permissible levels recommended by the FRC. The highest whole body dose equivalent received by an employee was about 4.4 rem or 37 percent of the maximum permissible annual dose. No employee has a cumulative whole body dose which exceeds the recommended maximum permissible dose as based on the age proration formula 5(N-18). There were no cases of internal exposure where the deposition of radioactive materials within the body was estimated to have averaged greater than one-half of a maximum permissible body burden.

During 1965, there were 41 unusual occurrences recorded, which is the second lowest number recorded since the present system of reporting unusual occurrences was established in 1960. The 41 occurrences is an increase of 42 percent over the 29 reported for 1964, the lowest number per year which has been reported, but is about 17 percent below the five-year average for the years 1961 through 1965.

The Laboratory reported 17 disabling injuries during 1965, which was a frequency rate of 2.2. The total number for the past five years (1961-1965) was 56, or an average frequency rate of 1.5.

ORNL-DWG 66-3988



Health Physics and Safety Organization Chart.

5.0 ENVIRONS MONITORING

The Health Physics Division monitors for airborne radioactivity in the East Tennessee area by the use of three separate monitoring networks. The local air monitoring (LAM) network consists of twenty-two stations which are positioned in relation to ORNL operational activities (Figures 1 and 2); the perimeter air monitoring (PAM) network consists of nine stations which are located on the perimeter of the AEC controlled area (Figure 3); and the remote air monitoring (RAM) network consists of eight stations which are located outside the AEC controlled area at distances of from 12 to 75 miles from ORNL (Figure 4). The monitoring networks provide for the collection of (1) airborne radioactivity by air filtration techniques, (2) radioparticulate fallout material by impingement on gummed paper trays, and (3) rain water for measurement of fallout occurring as rainout. The filter data are representative of radioparticulate matter which might be considered respirable; the gummed paper data are representative of radioparticulate fallout; and the rain water data provide information on the soluble and insoluble fractions of the radioactive content of fallout material.[1]

Low level radioactive liquid wastes originating from ORNL operations are discharged, after preliminary treatment, to White Oak Creek, which is a small tributary of the Clinch River. Liquid waste releases are controlled so that the resulting average radioactive concentrations in the Clinch River are well below the maximum permissible concentrations established for populations in the neighborhood of an atomic energy installation as recommended by the National Committee on Radiation Protection (NCRP) and the Federal Radiation Council (FRC).

The radioactive content of the White Oak Creek discharge is determined at White Oak Dam (Figure 5) which is the last control point along the stream prior to entry of White Oak Creek waters into Clinch River waters. Water samples are collected also at a number of locations along the Clinch River, beginning at a point above the entry of wastes into the river via White Oak Creek and ending at Center's Ferry (near Kingston, Tennessee) about 16 miles downstream from the confluence of White Oak Creek and the Clinch River. Water samples are analyzed for gross radioactivity and for certain specified long-lived radionuclides. A weighted average, $(MPC)_w$, for the mixture of radionuclides is calculated on the basis of the isotopic distribution in the water.

Samples of ORNL potable water are collected daily, composited and stored. At the end of each quarter these composites are analyzed radiochemically for ^{90}Sr content and are assayed for long-lived gamma emitting radionuclides by gamma spectrometry.

---

[1]A detailed discussion concerning techniques used in processing air and water samples for environmental monitoring purposes is given in ORNL-2601, "Radioactive Waste Management at Oak Ridge National Laboratory".

ORNL-DWG. 66-2218

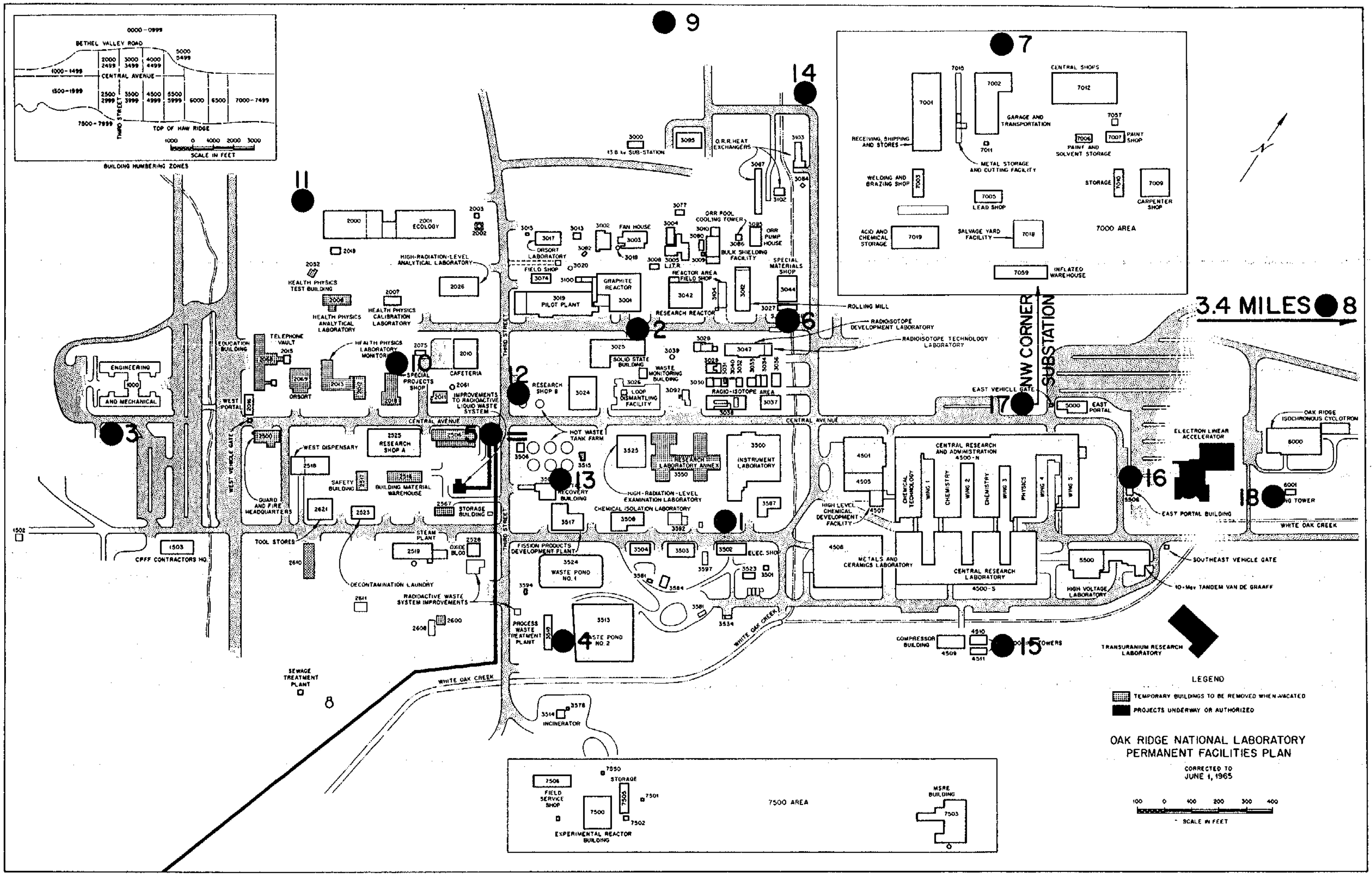


Fig. 1. Map of X-10 Area Showing the Approximate Location of 18 of 22 of the Local Monitoring Stations Constituting the LAM Network.

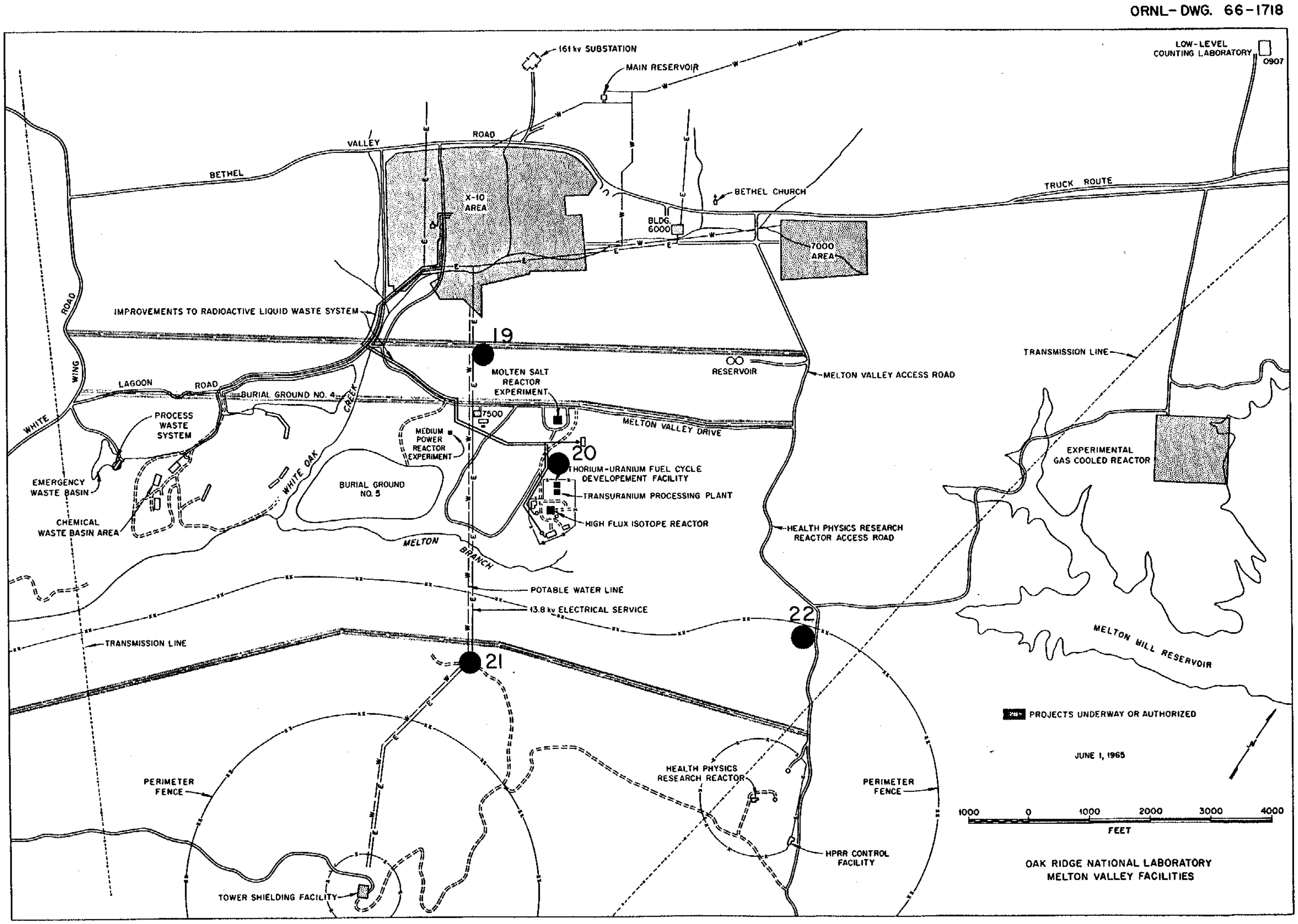


Fig. 2. Map of Laboratory Area Showing the Approximate Location of 4 of the 22 Local Monitoring Stations Constituting the LAM Network.

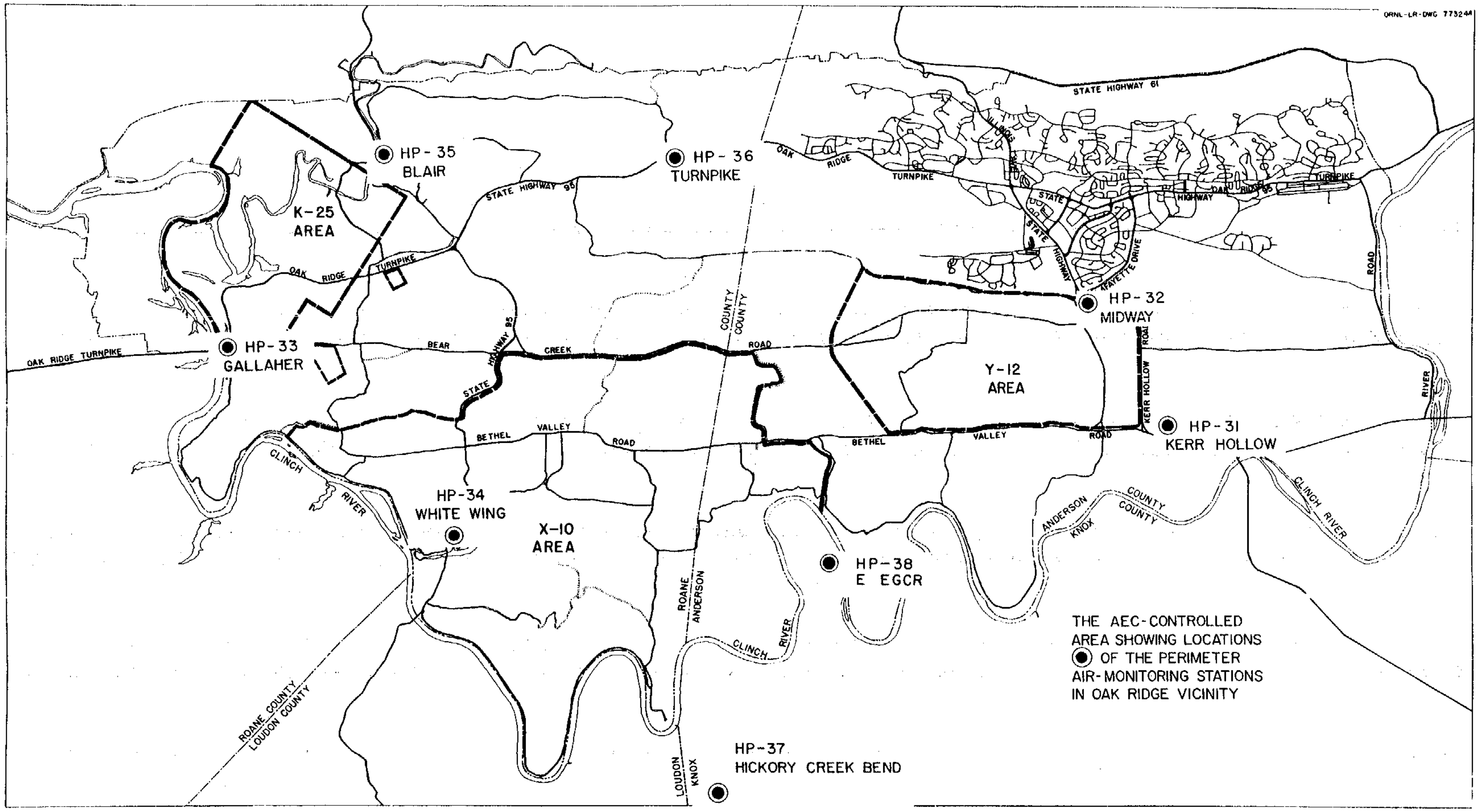


Fig. 3. Map of the AEC Controlled Area and Vicinity Showing the Approximate Location of the Perimeter Air Monitoring Stations Constituting the PAM Network.

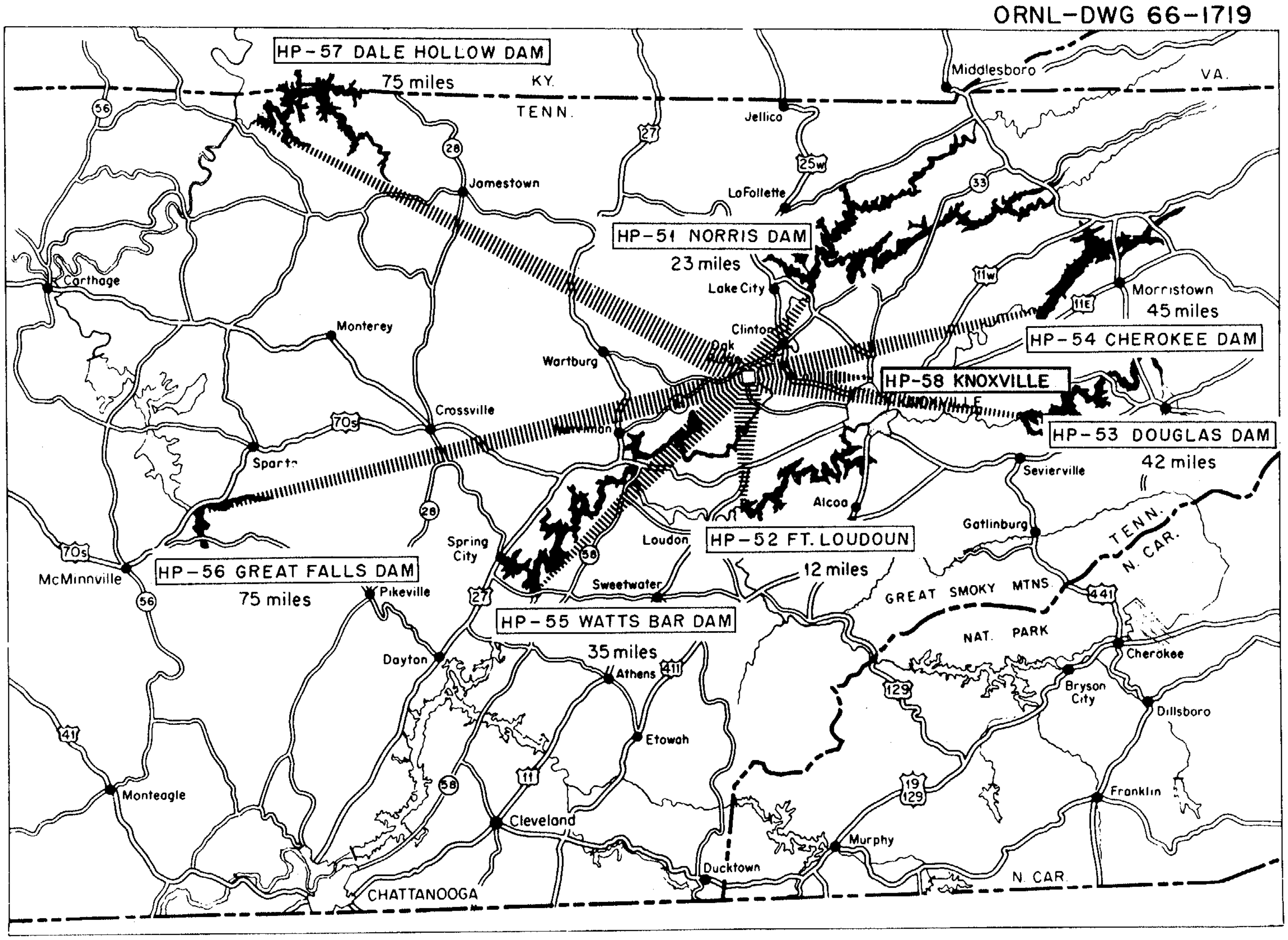


Fig. 4. Map of a Section of the East Tennessee Area Showing TVA and U.S. Corps of Engineering Dam Sites at Which Are Located the Remote Air Monitoring Stations Constituting the RAM Network.

ORNL-DWG. 66-2216

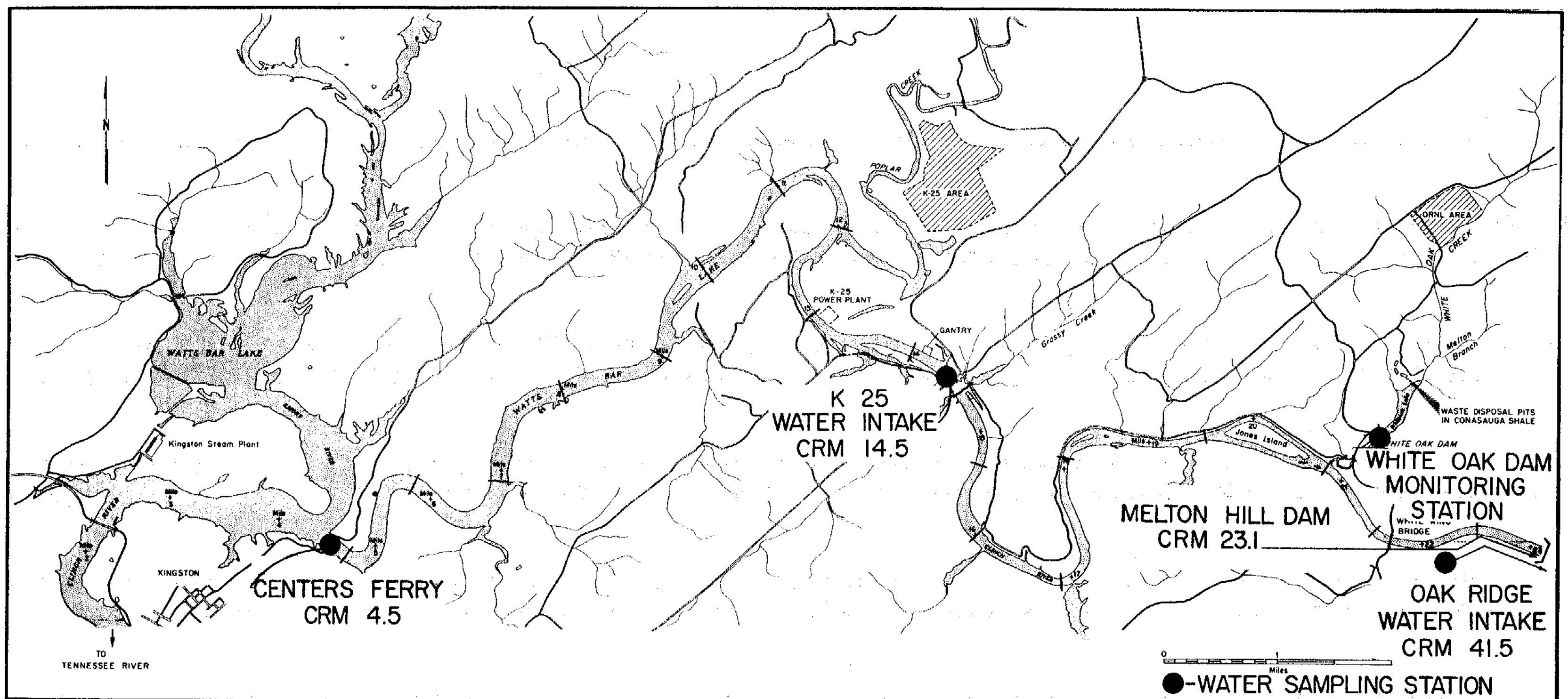


Fig. 5. Map Showing Water Sampling Locations in the East Tennessee Area.

Raw milk samples are collected at twelve sampling stations located within a radius of 50 miles from ORNL. Samples are taken on a weekly basis from eight stations which are located outside the AEC controlled area within a 12-mile radius of ORNL (Figure 6). Samples are collected every five weeks from the four remaining stations, all of which are located outside the 12-mile radius up to distances of about 50 miles. The purpose of the milk sampling program is twofold: first, samples collected in the immediate vicinity of ORNL provide data by which one may evaluate the possible effect of waste releases originating from ORNL operations; second, samples collected remote to the immediate vicinity of the ORNL area provide background data which are essential in establishing a proper index from which the intentional or accidental release of radioactive materials originating from Oak Ridge operations may be evaluated.

Thyroid tissues taken from cattle pastured within a radius of 100 miles of Oak Ridge are analyzed for radioiodine at the rate of six samples per week. These analyses provide information on background levels needed to identify environmental levels that might result from either continuous or sporadic releases of ^{131}I to the environment from ORNL and other Oak Ridge operations.

Aerial background surveys continue to be made over the ORNL area and for several miles from ORNL in the general direction of low altitude prevailing winds but the frequency of flights have been reduced to not more than once per quarter.

Background gamma radiation measurements are made monthly at a number of locations throughout other portions of the East Tennessee area. These measurements are taken with calibrated GM and scintillation type detectors at a distance of three feet above the surface of the ground.

River bottom sediments in the Clinch and Tennessee Rivers have been surveyed and analyzed annually since the year 1951 for the purpose of providing data relative to the dispersion of radioactive wastes released from Oak Ridge operations to the Clinch River.

5.1 Atmospheric Monitoring

One new remote air monitoring station was added to the network in 1965. Station No. 58 was installed at the TVA power service center in northeast Knoxville near Fountain City.

5.1.1 Air Concentrations - The average concentrations of radioactive materials in the atmosphere, as measured by filtration methods provided by the LAM, PAM, and RAM networks during 1965, were as follows:

Network	Concentration (μc/cc)
LAM	0.28×10^{-12}
PAM	0.21×10^{-12}
RAM	0.19×10^{-12}

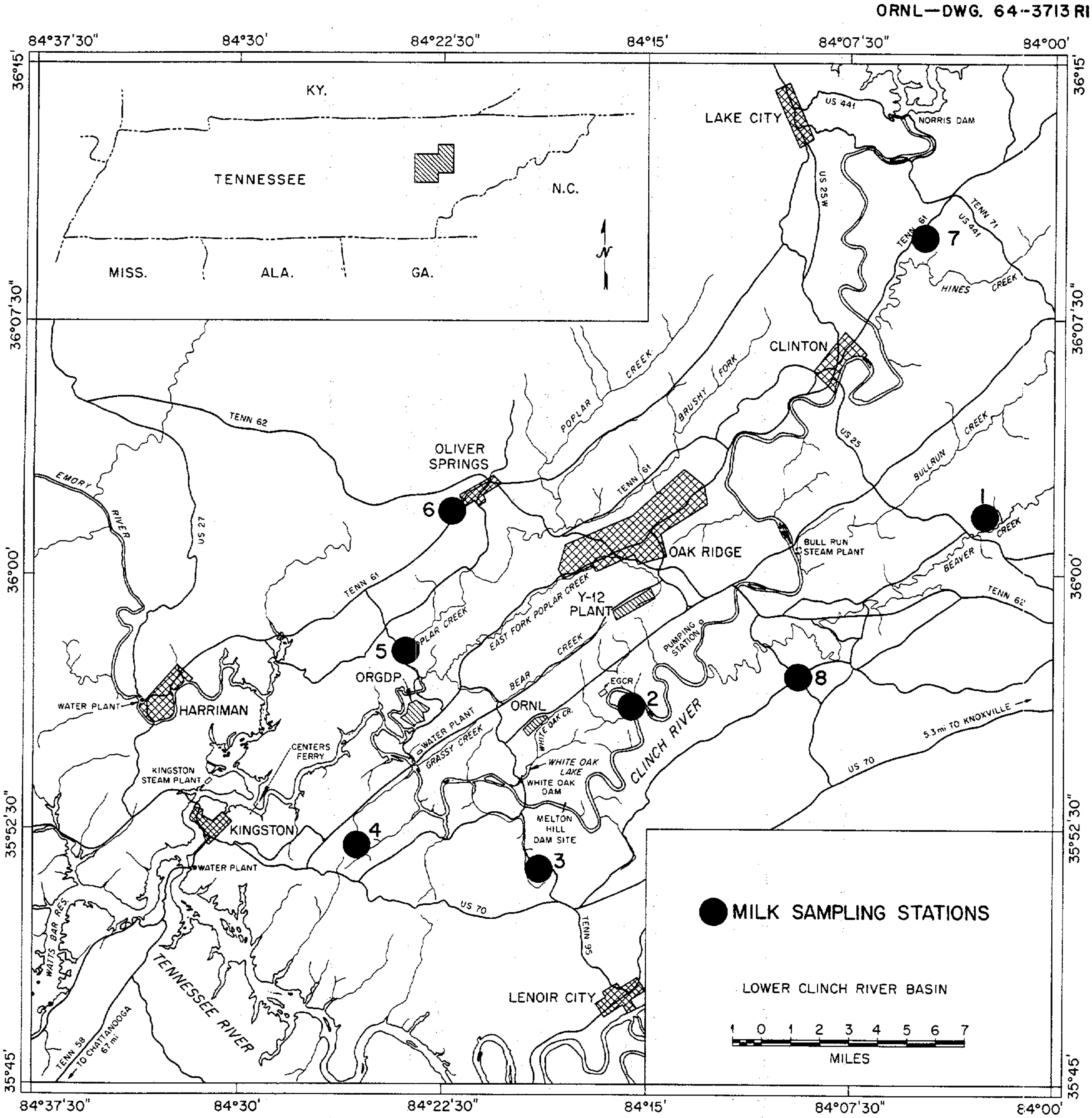


Fig. 6. Map Showing Milk Sampling Stations in the East Tennessee.

The LAM network value of 0.28 x 10^{-12} $\mu c/cc$ is about 0.03 percent of the $(MPCU)_a$[2] based on occupational exposure. The PAM and RAM network values represent 0.21 and 0.19 percent, respectively, of the $(MPCU)_a$ for persons residing in the neighborhood of an atomic energy installation. A tabulation of data for each station in each network is given in Table 1. The weekly values for each network are illustrated in Figure 7.

The number of radioactive particles collected on air monitor filters of the LAM network in 1965 decreased by a factor of 2 from the number collected in 1964. The values measured by the PAM and RAM networks remained approximately the same as in 1964. The average number of radioparticulates per 1000 cubic feet of air sampled at each station in each network is given in Table 1.

5.1.2 Fallout (Gummed Paper Technique) - Radioparticulate fallout as measured by the LAM network of stations increased by a factor of about 2 from the value measured in 1964. The values measured by both the PAM and RAM networks increased by a factor of approximately 7 from the 1964 values. The increases may be attributed to world wide fallout from weapons testing. The peak of the fallout occurred in the Oak Ridge area during the week ending May 24, 1965. Laboratory analysis of the fallout material confirmed the presence of fresh fission products, the age of which was consistent with the timing of the announced detonation on the Chinese mainland on May 14, 1965.[3] Table 2 gives a tabulation of data for each station within each network. The weekly average values for each network for each week are illustrated in Figure 8.

5.1.3 Atmospheric Radioiodine (Charcoal Collector Techniques) - Atmospheric radioiodine measured by the perimeter stations averaged 0.015 x 10^{-12} $\mu c/cc$ during 1964. This is only about 0.02 percent of the maximum permissible concentration for populations in the neighborhood of a controlled area. The maximum value observed at any one station for one week was 0.20 x 10^{-12} $\mu c/cc$. This value was measured at the White Oak Dam station and was associated with the release of about 2.7 curies of radioiodine from ORNL[4] stacks during a period of one week. Figure 9 compares the weekly discharge of radioiodine from ORNL stacks[5] with the average concentration of radioiodine measured by the perimeter stations.

---

[2]The $(MPCU)_a$ is defined as the maximum permissible concentration for an unknown mixture of radioisotopes in air. NBS Handbook 69, Table 4, p. 94, gives exposure values applicable to various mixtures of radionuclides and establishes guide lines for deriving the $(MPCU)_a$.

[3]Radiological Health Data, Volume 6, Number 6, June, 1965, U. S. Department of Health, Education and Welfare.

[4]"Summary of Waste Discharges", Week Ending January 17, 1965, L. C. Lasher.

[5]"Summary of Waste Discharges", Weekly Reports, 1965, L. C. Lasher.

If 22, why only 10 here?

Table 1 CONCENTRATION OF RADIOACTIVE MATERIALS IN AIR - 1965
(Filter Paper Data—Weekly Average)

Station Number	Location	Long-Lived Activity 10^{-13} μc/cc	No. of Particles by Activity Ranges					Particles Per 1000 ft^3
			< 10^5 d/24 hr	10^5-10^6 d/24 hr	10^6-10^7 d/24 hr	> 10^7 d/24 hr	Total	
Laboratory Area								
HP-1	S 3587	2.6	5.3	0.00	0.00	0.00	5.3	0.22
HP-2	NE 3025	3.6	0.76	0.04	0.00	0.00	0.80	0.04
HP-3	SW 1000	2.6	0.81	0.00	0.00	0.00	0.81	0.04
HP-4	W Settling Basin	3.2	0.84	0.00	0.00	0.00	0.84	0.05
HP-5	E 2506	3.3	1.2	0.00	0.00	0.00	1.2	0.07
HP-6	SW 3027	2.7	0.56	0.02	0.00	0.00	0.58	0.03
HP-7	W 7001	1.9	0.02	0.00	0.00	0.00	0.02	0.00
HP-8	Rock Quarry	2.2	0.57	0.00	0.00	0.00	0.57	0.03
HP-9	N Bethel Valley Rd.	2.5	0.98	0.00	0.00	0.00	0.98	0.05
HP-10	W 2075	2.9	0.54	0.02	0.00	0.00	0.56	0.03
Average		2.8	1.2	0.01	0.00	0.00	1.2	0.06
Perimeter Area								
HP-31	Kerr Hollow Gate	1.7	1.1	0.00	0.00	0.00	1.1	0.02
HP-32	Midway Gate	1.9	1.2	0.02	0.00	0.00	1.2	0.02
HP-33	Gallaher Gate	1.5	0.75	0.00	0.00	0.00	0.75	0.02
HP-34	White Wing Gate	1.8	0.71	0.00	0.00	0.00	0.71	0.02
HP-35	Blair Gate	1.9	1.8	0.00	0.00	0.00	1.8	0.04
HP-36	Turnpike Gate	2.2	0.63	0.02	0.00	0.00	0.65	0.01
HP-37	Hickory Creek Bend	1.6	1.0	0.00	0.00	0.00	1.0	0.02
HP-38	E EGCR	2.2	1.4	0.00	0.00	0.00	1.4	0.04
HP-39	Townsite	3.5	1.7	0.00	0.00	0.00	0.0	0.05
Average		2.1	1.1	0.00	0.00	0.00	1.2	0.03
Remote Area								
HP-51	Norris Dam	2.1	1.2	0.04	0.00	0.00	1.2	0.02
HP-52	Loudoun Dam	1.7	0.98	0.02	0.00	0.00	1.0	0.02
HP-53	Douglas Dam	1.9	0.77	0.00	0.00	0.00	0.77	0.02
HP-54	Cherokee Dam	1.9	1.2	0.02	0.00	0.00	1.3	0.02
HP-55	Watts Bar Dam	1.9	1.5	0.02	0.00	0.00	1.5	0.03
HP-56	Great Falls Dam	1.8	1.4	0.02	0.00	0.00	1.4	0.03
HP-57	Dale Hollow Dam	1.8	1.3	0.02	0.00	0.00	1.3	0.03
*HP-58	Knoxville	2.0	1.8	0.05	0.00	0.00	1.9	0.04
Average		1.9	1.3	0.02	0.00	0.00	1.3	0.02

* Installed March, 1965.

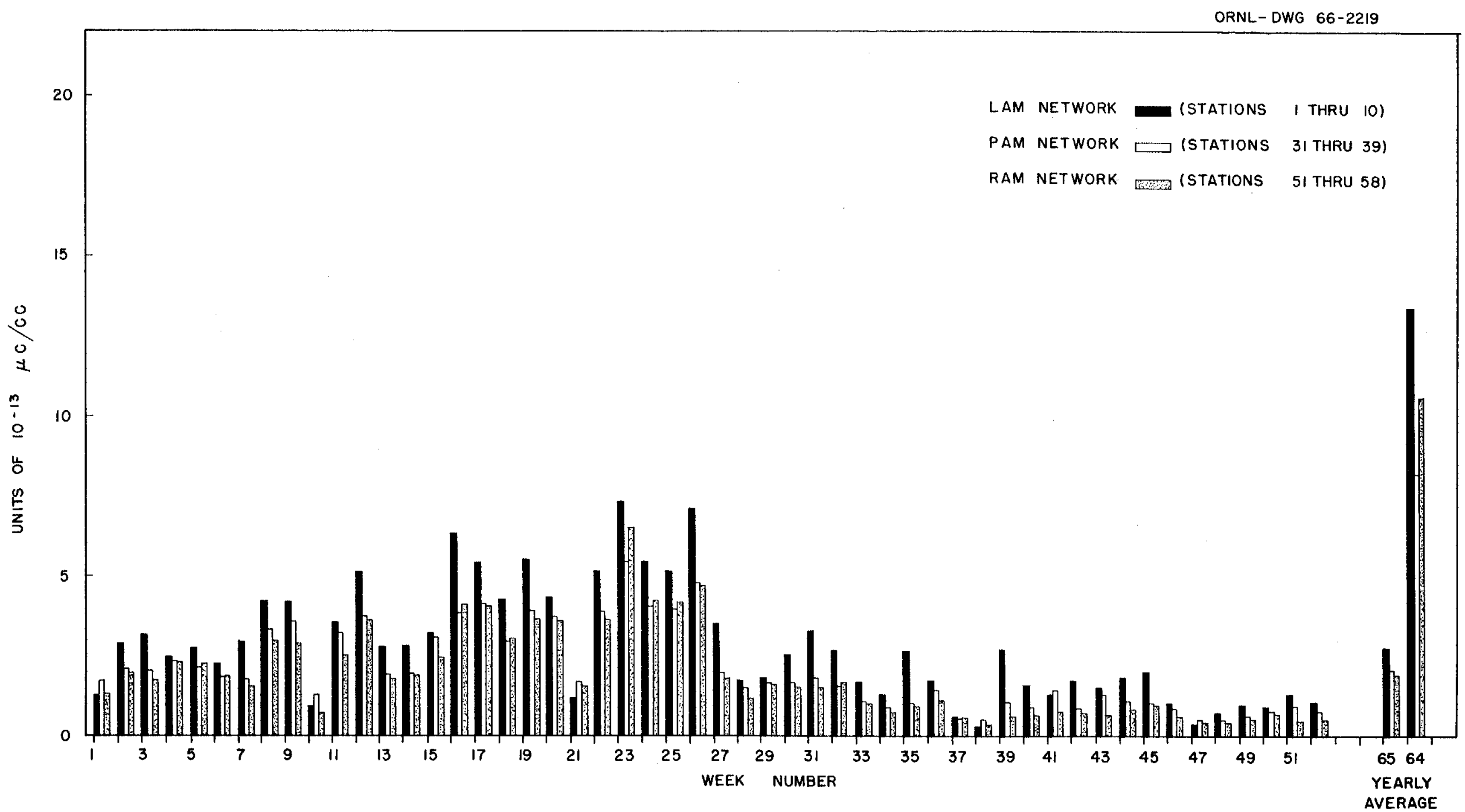


Fig. 7. Concentration of Radioactive Materials in Air as Determined from Filter Paper Data - 1965.

Table 2 RADIOPARTICULATE FALLOUT —1965
(Gummed Paper Data—Weekly Average)

Station Number	Location	Long-Lived Activity 10^{-4} μc/ft^2	No. of Particles by Activity Ranges				Total Particles Per Sq. Ft.
			< 10^5 d/24 hr	10^5-10^6 d/24 hr	10^6-10^7 d/24 hr	> 10^7 d/24 hr	
			Laboratory Area				
HP-1	S 3587	1.3	1.71	0.15	0.00	0.00	1.87
HP-2	NE 3025	1.1	1.61	0.06	0.04	0.02	1.73
HP-3	SW 1000	0.7	1.13	0.08	0.00	0.00	1.21
HP-4	W Settling Basin	1.4	0.67	0.04	0.00	0.00	0.71
HP-5	E 2506	0.6	1.23	0.19	0.00	0.00	1.42
HP-6	SW 3027	1.1	1.85	0.13	0.02	0.00	2.00
HP-7	W 7001	0.4	0.20	0.00	0.00	0.00	0.20
HP-8	Rock Quarry	0.5	0.83	0.06	0.00	0.02	0.91
HP-9	N Bethel Valley Rd.	0.4	1.00	0.06	0.00	0.00	1.06
HP-10	W 2075	2.1	2.29	0.17	0.00	0.04	2.50
Average		1.0	1.28	0.10	0.01	0.01	1.40
			Perimeter Area				
HP-31	Kerr Hollow Gate	0.48	0.23	0.13	0.00	0.00	0.37
HP-32	Midway Gate	0.56	1.12	0.00	0.00	0.00	1.12
HP-33	Gallaher Gate	0.41	0.92	0.06	0.00	0.00	0.98
HP-34	White Wing Gate	0.44	0.88	0.02	0.00	0.00	0.90
HP-35	Blair Gate	0.44	0.90	0.02	0.00	0.00	0.92
HP-36	Turnpike Gate	0.48	0.92	0.04	0.00	0.00	0.96
HP-37	Hickory Creek Bend	0.42	0.79	0.04	0.00	0.00	0.83
HP-38	E EGCR	0.45	1.25	0.02	0.00	0.00	1.27
HP-39	Townsite	0.49	1.42	0.02	0.00	0.00	1.44
Average		0.46	0.94	0.04	0.00	0.00	0.98
			Remote Area				
HP-51	Norris Dam	0.45	0.37	0.02	0.00	0.00	0.38
HP-52	Loudoun Dam	0.37	0.17	0.13	0.00	0.00	0.31
HP-53	Douglas Dam	0.36	0.40	0.04	0.00	0.00	0.44
HP-54	Cherokee Dam	0.35	0.46	0.02	0.00	0.00	0.48
HP-55	Watts Bar Dam	0.39	0.35	0.02	0.00	0.00	0.37
HP-56	Great Falls Dam	0.42	1.06	0.06	0.00	0.00	1.12
HP-57	Dale Hollow Dam	0.45	0.63	0.02	0.00	0.00	0.65
*HP-58	Knoxville	0.40	0.45	0.10	0.00	0.00	0.55
Average		0.40	0.49	0.05	0.00	0.00	0.54

*Installed March, 1965.

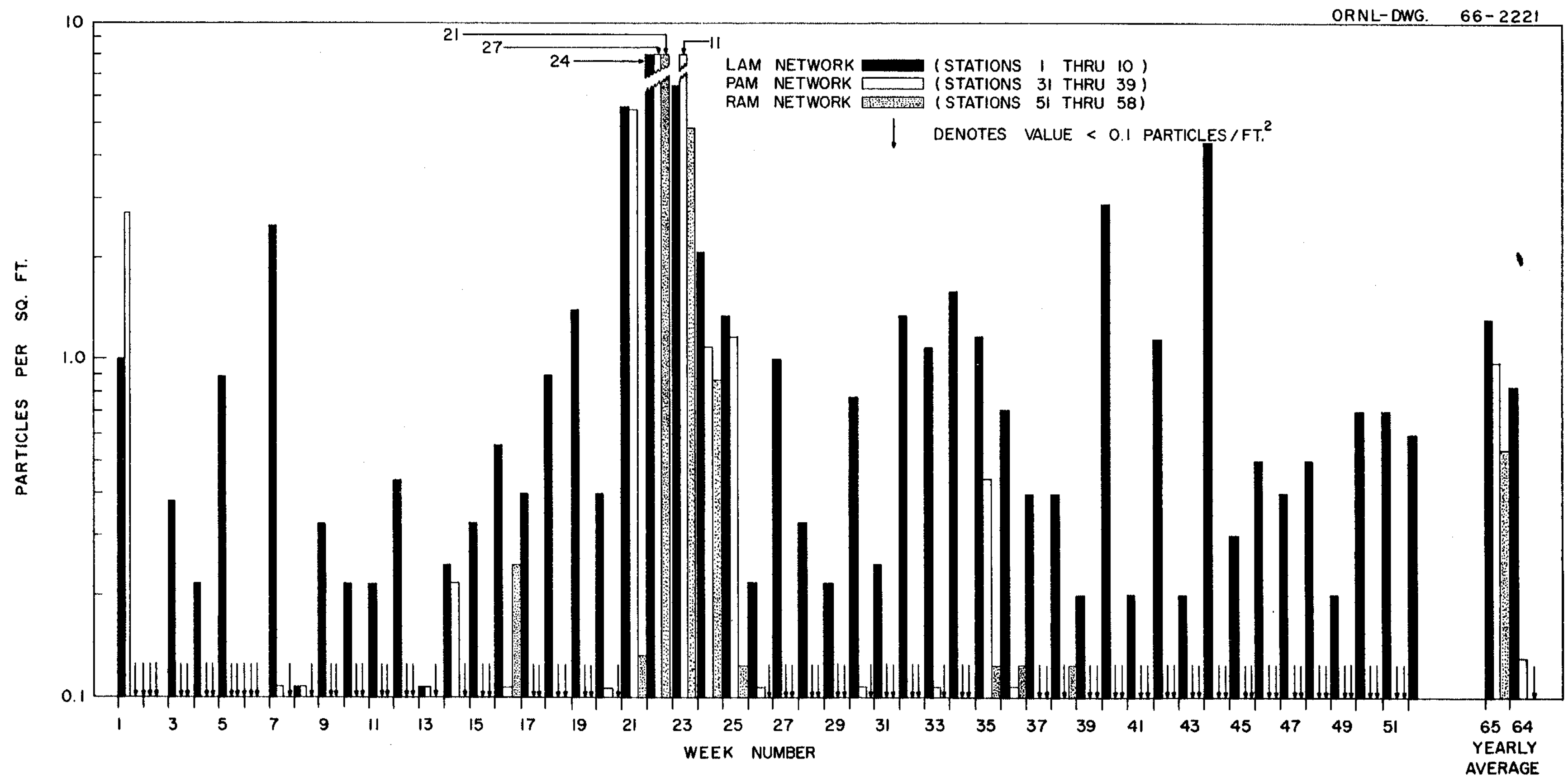


Fig. 8. Radioparticulate Fallout Measurements as Determined by Autoradiographic Techniques Using Gummed Paper Collectors - 1965.

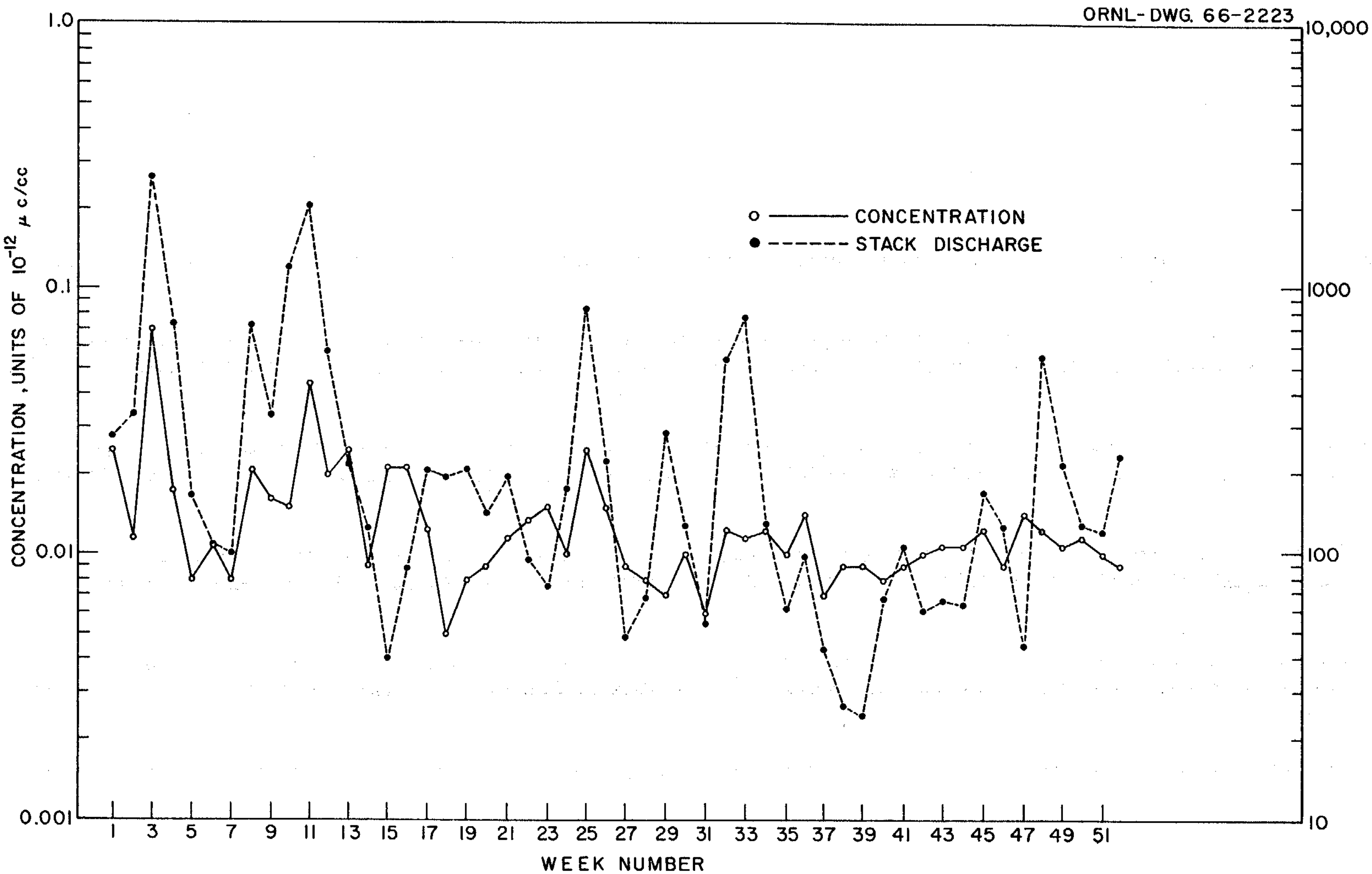


Fig. 9. Weekly Average Concentration of ^{131}I in Air at the Perimeter of the Controlled Area Compared with ^{131}I Discharges from ORNL Stacks - 1965.

5.2 Water Analyses

5.2.1 Rain Water - The average concentration of radioactivity in rain water collected from the three networks during 1965 were as follows:

Network	Concentration (μc/ml)
LAM	0.35×10^{-7}
PAM	0.42×10^{-7}
RAM	0.50×10^{-7}

These values are lower than those observed during 1964 by a factor of approximately 4 on the LAM network and by a factor of approximately 3.5 on both the PAM and RAM networks. The average values for each station are shown in Table 3; the average values for each network for each week are given in Figure 10.

5.2.2 Clinch River Water - A total of 95 beta curies of radioactivity was released to the Clinch River during 1965 as compared to 234 for 1964 (Table 4). Yearly discharges of radionuclides to Clinch River, 1949 through 1965, are shown in Table 5. Radiochemical analysis of the White Oak Dam effluent indicated that about 73 percent of the radioactivity was ^{106}Ru. The percentage of ^{90}Sr in the effluent was 3.6 compared to 2.8 in 1964.

The calculated average concentration of radioactive materials in the Clinch River at Clinch River Mile (CRM) 20.8 (the point of entry of White Oak Creek into the river) was 0.28×10^{-7} μc/ml. This represents only 0.61 percent of the weighted average $(MPC)_w$ recommended for persons residing in the neighborhood of an atomic energy installation (Table 6). The average concentration of radioactive materials in the Clinch River did not exceed 3.4 percent of the $(MPC)_w$ during any week in 1965 (Figure 11).

The measured average concentration of radioactivity in Clinch River water at CRM 41.5 (above the entry of White Oak Creek) was 0.23 percent of the weighted average $(MPC)_w$ (Table 6). The concentration of ^{90}Sr in the river above the entry of White Oak Creek continues to be about the same as the contribution calculated for White Oak Creek effluent at CRM 20.8 assuming uniform mixing of the two streams.

The measured average concentration of radioactive materials in the Clinch River at CRM 4.5 (near Kingston, Tennessee) was 1.7×10^{-8} μc/ml. This value represents 0.60 percent of the $(MPC)_w$ as applied to persons living in the neighborhood of an atomic energy installation.

5.2.3 Potable Water - The average concentrations of ^{90}Sr in potable water at ORNL during 1965 were as follows:

Table 3 CONCENTRATION OF RADIOACTIVE MATERIALS IN RAINWATER - 1965
(Weekly Average by Stations)

Station Number	Location	Activity in Collected Rainwater, μc/ml
	Laboratory Area	
HP-7	West 7001	0.35×10^{-7} μc/ml
	Perimeter Area	
HP-31	Kerr Hollow Gate	0.40×10^{-7} μc/ml
HP-32	Midway Gate	0.42
HP-33	Gallaher Gate	0.42
HP-34	White Wing Gate	0.38
HP-35	Blair Gate	0.48
HP-36	Turnpike Gate	0.37
HP-37	Hickory Creek Bend	0.40
HP-38	E EGCR	0.59
HP-39	Townsite	0.39
Average		0.42×10^{-7} μc/ml
	Remote Area	
HP-51	Norris Dam	0.62×10^{-7} μc/ml
HP-52	Loudoun Dam	0.58
HP-53	Douglas Dam	0.49
HP-54	Cherokee Dam	0.57
HP-55	Watts Bar Dam	0.42
HP-56	Great Falls Dam	0.47
HP-57	Dale Hollow Dam	0.49
HP-58	Knoxville	0.40
Average		0.50×10^{-7} μc/ml

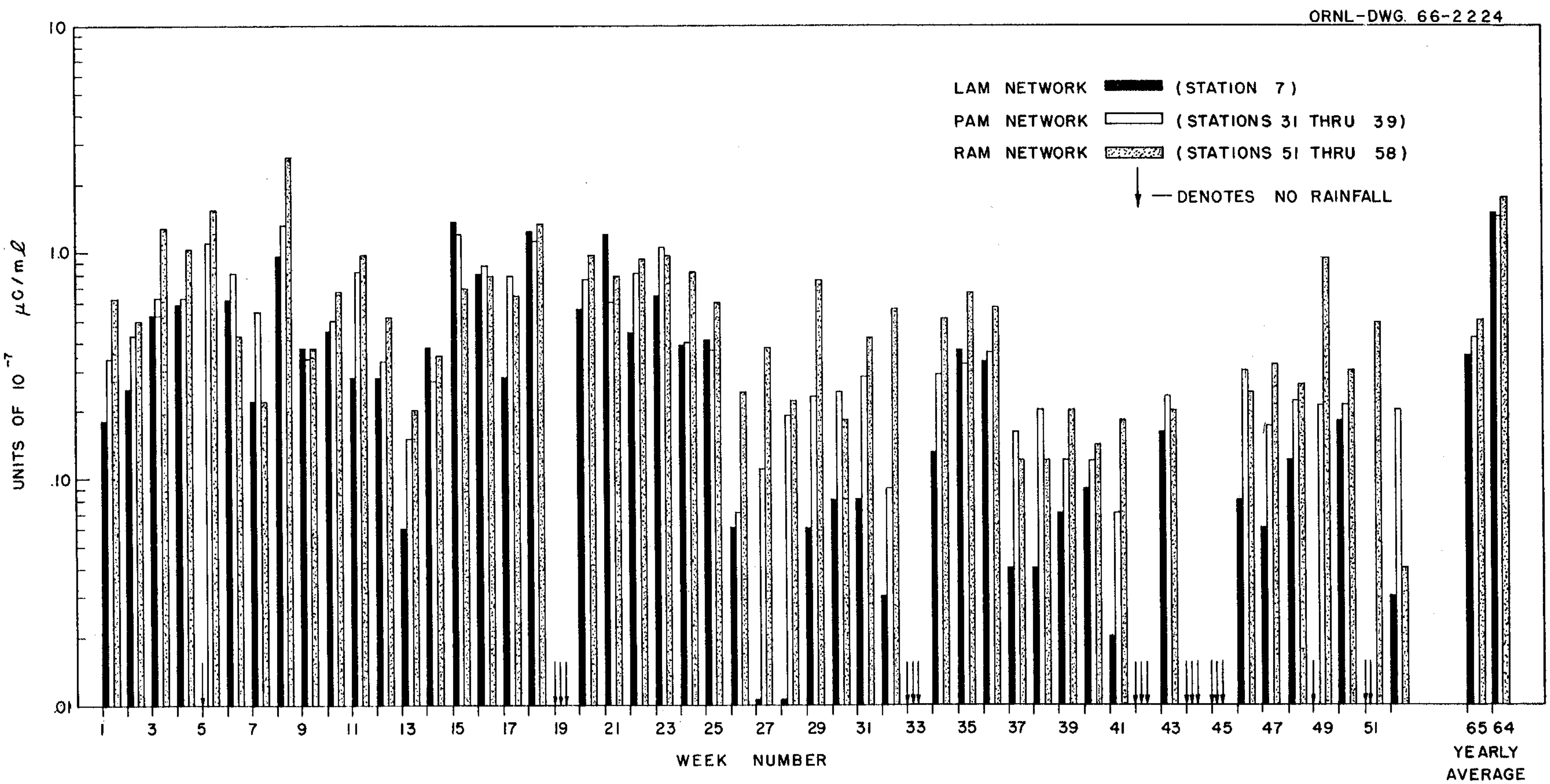


Fig. 10. Concentration of Radioactive Materials in Rainwater - 1965.

Table 4 LIQUID WASTES DISCHARGED FROM WHITE OAK CREEK - 1965

	Curies	
	Total for Year	Weekly Average
Beta Activity	95	1.8
Transuranic Alpha Emitters	0.50	0.010

Table 5 YEARLY DISCHARGES OF RADIONUCLIDES TO CLINCH RIVER (CURIES)

Year	Gross Beta	^{137}Cs	^{106}Ru	^{90}Sr	TRE (-Ce)	^{144}Ce	^{95}Zr	^{95}Nb	^{131}I	^{60}Co
1949	718	77	110	150	77	18	180	22	77	
1950	191	19	23	38	30		15	42	19	
1951	101	20	18	29	11		4.5	2.2	18	
1952	214	9.9	15	72	26	23	19	18	20	
1953	304	6.4	26	130	110	6.7	7.6	3.6	2.1	
1954	384	22	11	140	160	24	14	9.2	3.5	
1955	437	63	31	93	150	85	5.2	5.7	7.0	6.6
1956	582	170	29	100	140	59	12	15	3.5	46
1957	397	89	60	83	110	13	23	7.1	1.2	4.8
1958	544	55	42	150	240	30	6.0	6.0	8.2	8.7
1959	937	76	520	60	94	48	27	30	0.5	77
1960	2190	31	1900	28	48	27	38	45	5.3	72
1961	2230	15	2000	22	24	4.2	20	70	3.7	31
1962	1440	5.6	1400	9.4	11	1.2	2.2	7.7	0.36	14
1963	470	3.5	430	7.8	9.4	1.5	0.34	0.71	0.44	14
1964	234	6.0	191	6.6	13	0.3	0.16	0.07	0.29	15
1965	95	2.1	69	3.4	5.9	0.1	0.33	0.33	0.20	12

Table 6 RADIOACTIVITY IN CLINCH RIVER - 1965

Location	Concentration of Radionuclides of Primary Concern in Units of 10^{-8} μc/ml						Average Concentration of Total Radioactivity	$(MPC)_w$ [a]	% of $(MPC)_w$
	^{90}Sr	^{144}Ce	^{137}Cs	$^{103-106}Ru$	^{60}Co	$^{95}Zr-^{95}Nb$	10^{-8} μc/ml	10^{-6} μc/ml	
CRM 41.5[b]	0.06	0.03	0.03	0.24	*	*	0.36	1.6	0.23
CRM 20.8[c]	0.06	0.01	0.03	0.79	0.17	< 0.01	2.8	4.6	0.61
CRM 4.5[b]	0.15	0.04	0.17	1.2	0.23	0.01	1.7	2.8	0.60

[a]Weighted average $(MPC)_w$ calculated for the mixture, using $(MPC)_w$ values for specific radionuclides specified by AEC Manual, Chapter 0524, Appendix, Annex 1, Table II.

[b]Measured values.

[c]Values given for this location are calculated values based on the levels of waste released and the dilution afforded by the river; they do not include amounts of radioactive material (e.g., fallout) that may enter the river upstream from CRM 20.8.

*None detected.

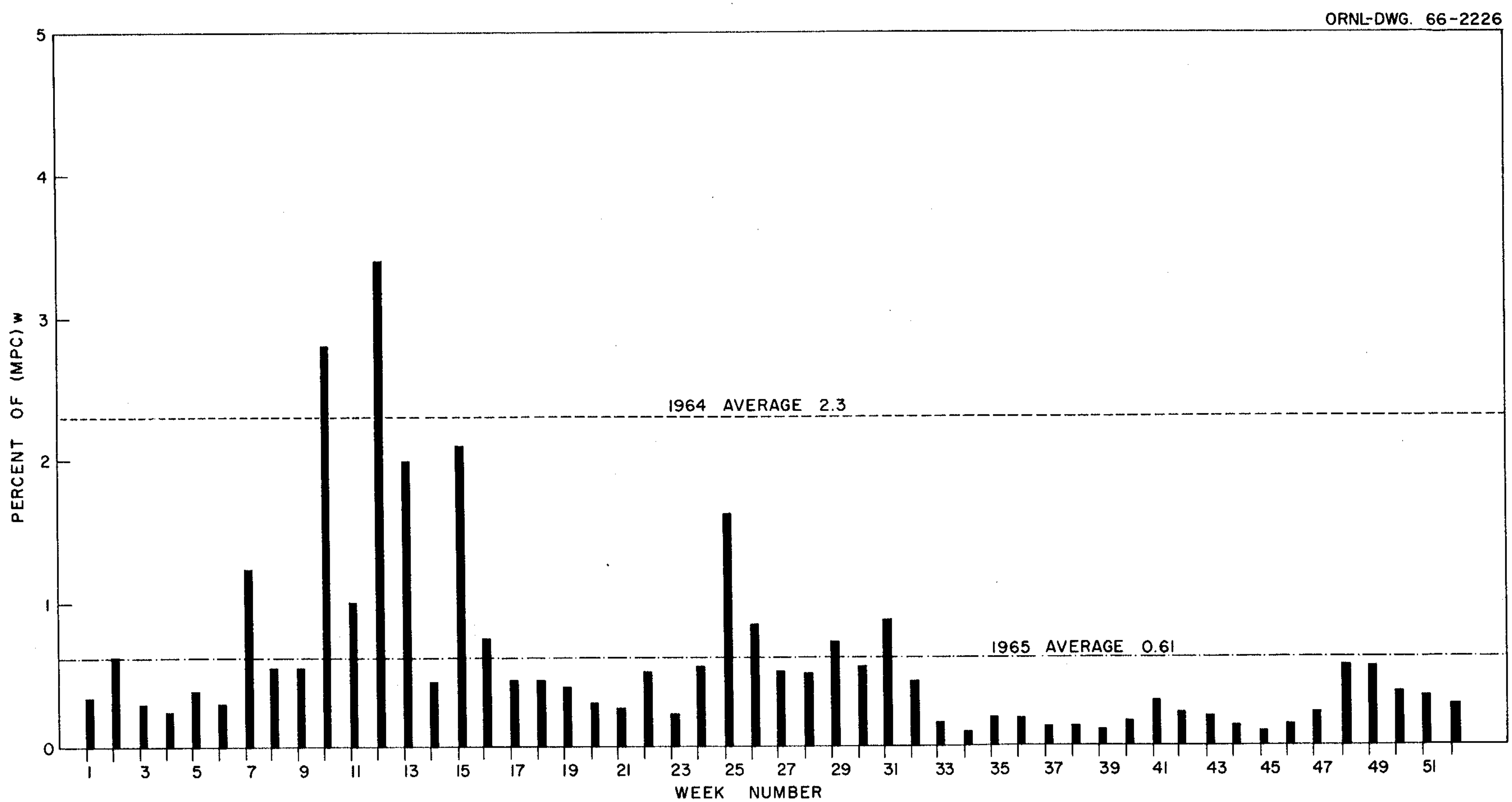


Fig. 11. Estimated Percent $(MPC)_W$ of Radioactivity in Clinch River Water Below the Mouth of White Oak Creek - 1965.

Quarter Number	Concentration ^{90}Sr (μc/ml)
1	0.79×10^{-9}
2	0.45×10^{-9}
3	0.45×10^{-9}
4	0.90×10^{-9}
Average for Year	0.64×10^{-9}

The average value of 0.64×10^{-9} represents 0.21 percent of the $(MPC)_w$ as applied to persons residing in the neighborhood of an atomic energy installation.

Based on gamma spectrometric analyses, no long-lived gamma emitting radionuclides were detected in ORNL potable water during 1965.

5.3 Milk Analyses

The average concentration of ^{90}Sr in raw milk samples collected within a 12-mile radius of the Laboratory during 1965 was 19.8 pc/l. The average concentration of ^{90}Sr in samples collected between 12 miles and 50 miles from the Laboratory was 20.4 pc/l. These results would indicate that the ^{90}Sr content of milk in the Oak Ridge area is largely the result of fallout from previous world wide weapons tests. Figure 12 presents the weekly average concentration of ^{90}Sr in raw milk sampled from the immediate environs of Oak Ridge.

The average concentration of ^{131}I in raw milk samples collected within a 12-mile radius of the Laboratory during 1965 was 7.4 pc/l. Figure 13 presents the weekly average concentrations of ^{131}I in raw milk collected at these stations compared with the weekly discharges of ^{131}I from the ORNL stacks. The peak concentration occurred during week 23 and may be attributed to the announced nuclear detonation on the Chinese mainland on May 14, 1965.[6] It should be noted that the yearly average concentration is below the lower limit of FRC Range II daily intake guide for ^{131}I, if one assumes an intake of 1 liter of milk per day, and that at no time during the year did the weekly average concentration exceed the upper limit of FRC Range II.

The radioiodine content of cattle thyroids average 18 pc/g of tissue during 1965. The iodine concentration reached a maximum during the month of June, just following the test on the Chinese mainland,[6] and had returned to pretest levels by October. The average ^{131}I content of thyroids for the month of June was 154 pc/g and the highest single sample measured contained 340 pc/g. The average monthly concentration of radioiodine in cattle thyroids for 1965 is given in Figure 14.

---

[6] Radiological Health Data, Volume 6, Number 6, June, 1965, U.S. Department of Health, Education and Welfare.

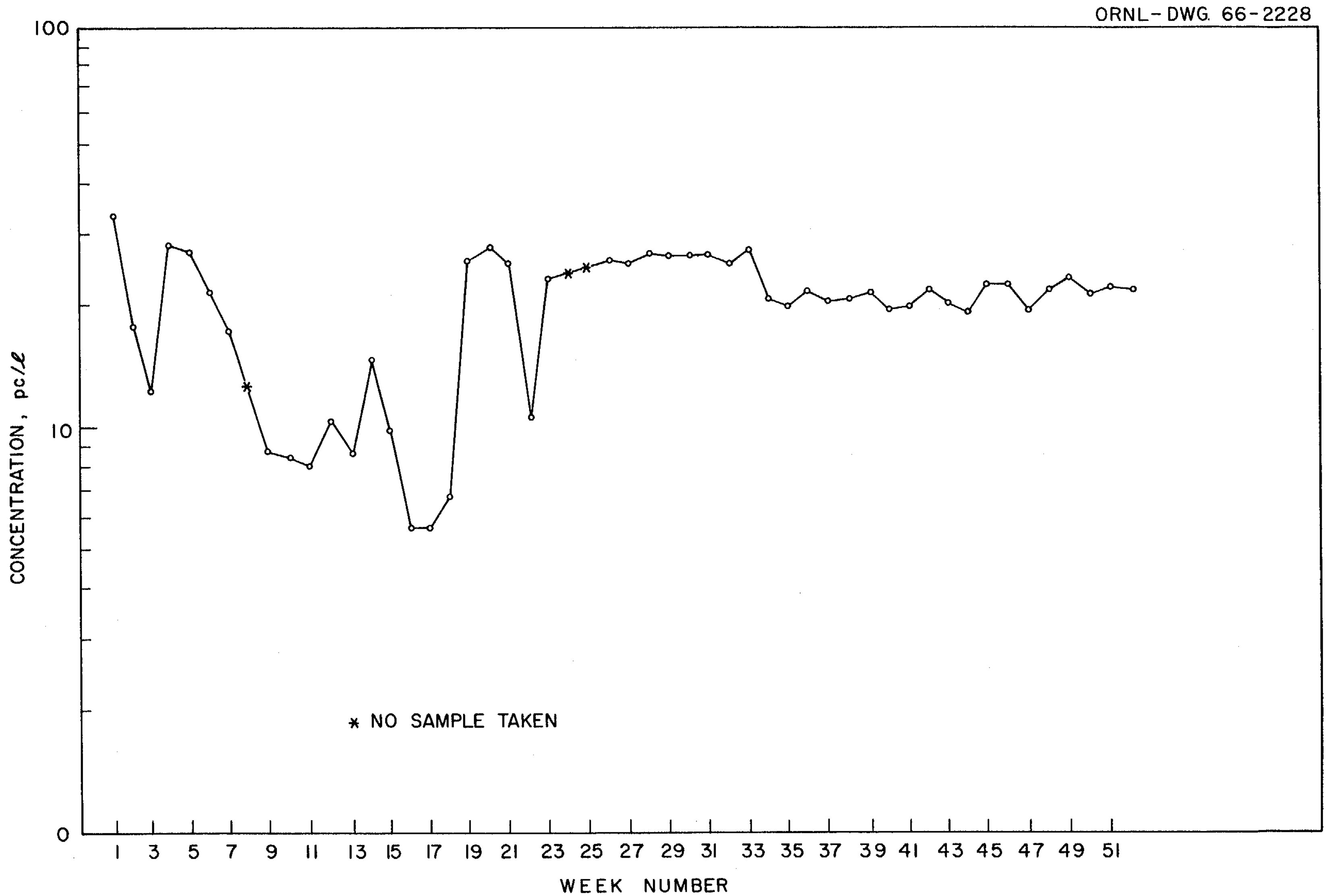


Fig. 12. Weekly Average Concentration of ^{90}Sr in Raw Milk in the Immediate Environs of Oak Ridge - 1965.

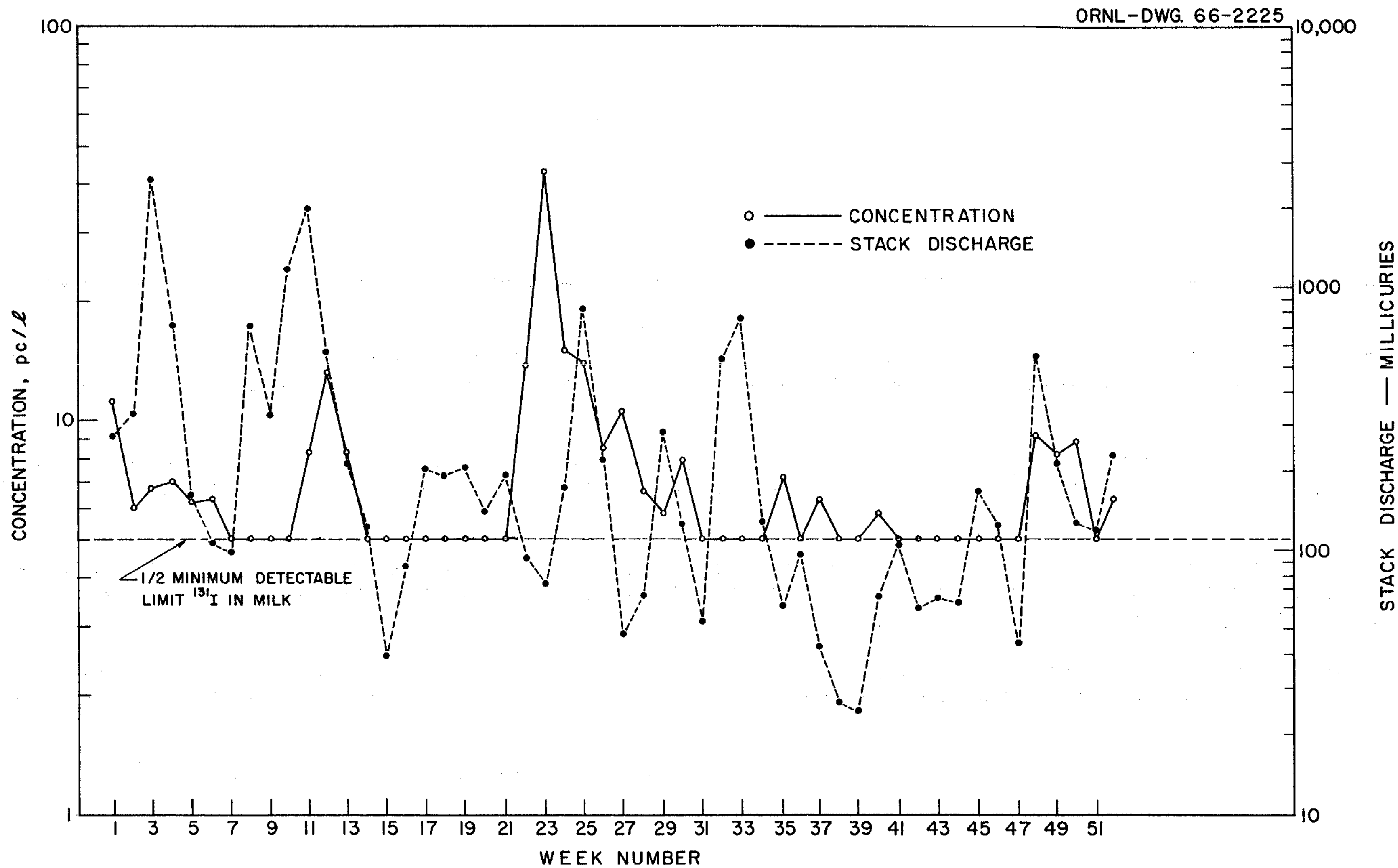


Fig. 13. Weekly Average Concentration of ^{131}I in Raw Milk in the Immediate Environs of Oak Ridge Compared with ^{131}I Discharges from ORNL Stacks - 1965.

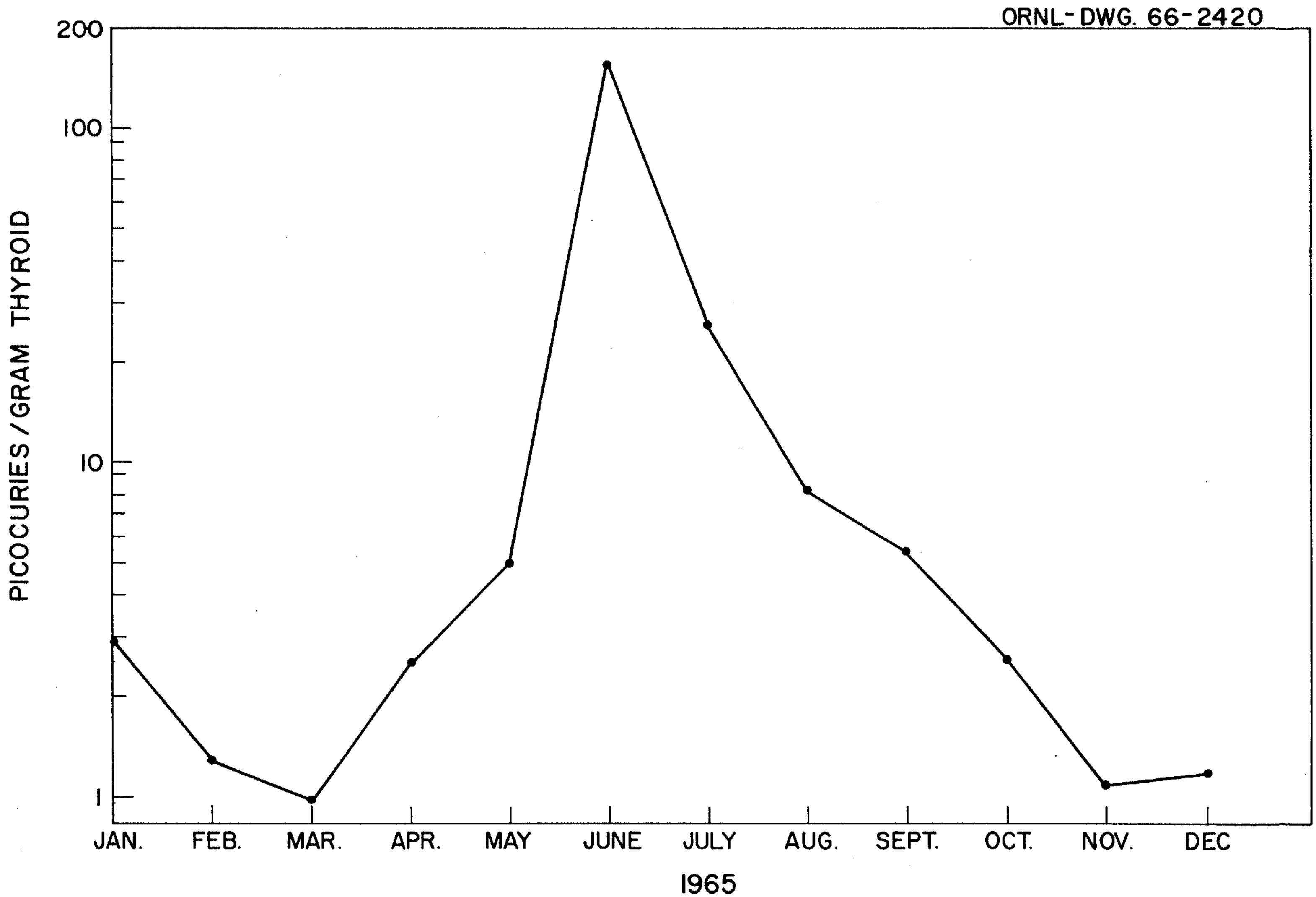


Fig. 14. Concentration of ^{131}I in Cattle Thyroids - 1965.

5.4 Background Measurements

Background measurements were taken at a number of locations (established in 1961) in the East Tennessee area during routine servicing visits to the remote air monitoring stations. Measurements were made at each location on a frequency of once each five weeks. The average background level during 1965 as measured at these stations was 0.013 mR/hr. Average background readings and the location of each station are presented in Figure 15.

Background measurements made monthly with a calibrated GM monitor at five selected locations adjacent to the ORNL area yielded an average background reading of 0.012 mR/hr during 1965. Corresponding measurements made at 53 locations on the ORNL site gave an average background of 0.072 mR/hr. The average background level measured in the Oak Ridge area in 1943 prior to the start-up of the Oak Ridge Graphite Reactor was 0.012 mR/hr. A comparison of average background values taken both on and off the X-10 site for the years 1950-65 is presented in Figure 16.

5.5 Annual Survey of the Clinch and Tennessee Rivers

The 1965 annual survey of the Clinch and Tennessee Rivers was curtailed in accordance with the recommendations of the Clinch River Study Steering Committee. Nineteen traverses were made in the Clinch and Tennessee Rivers in 1965 as compared to 36 in 1964. The same expanse of the Clinch River was covered as before but with fewer traverses being made. The survey in the Tennessee River extended downstream only as far as Watts Bar Dam. The techniques and procedures used are described in ORNL-2847, "Radioactivity in Silt of the Clinch and Tennessee Rivers".

The 1965 survey showed the dispersal pattern of radioactive silt in the Clinch River to be essentially the same as in 1964 but the levels of radioactivity measured were smaller (Figure 17). The average of the levels measured in 1965 cannot be compared directly to the average of all measurements made in previous surveys. The measurements made in 1965 were at or near points of previous maximum radioactive silt accumulation and hence would give a somewhat higher average than would have been the case if both "high" and "low" points of silt accumulation had been measured as was done in previous surveys.

The average gamma count rate on bottom silt located in Melton Hill Reservoir on the Clinch River and in Watts Bar Reservoir on the Tennessee River remained essentially the same as in 1964 (Figure 18 and Figure 19).

Radiochemical analysis data obtained from the Clinch and Tennessee River silt collected in the 1964 and 1965 surveys are given in Table 7.

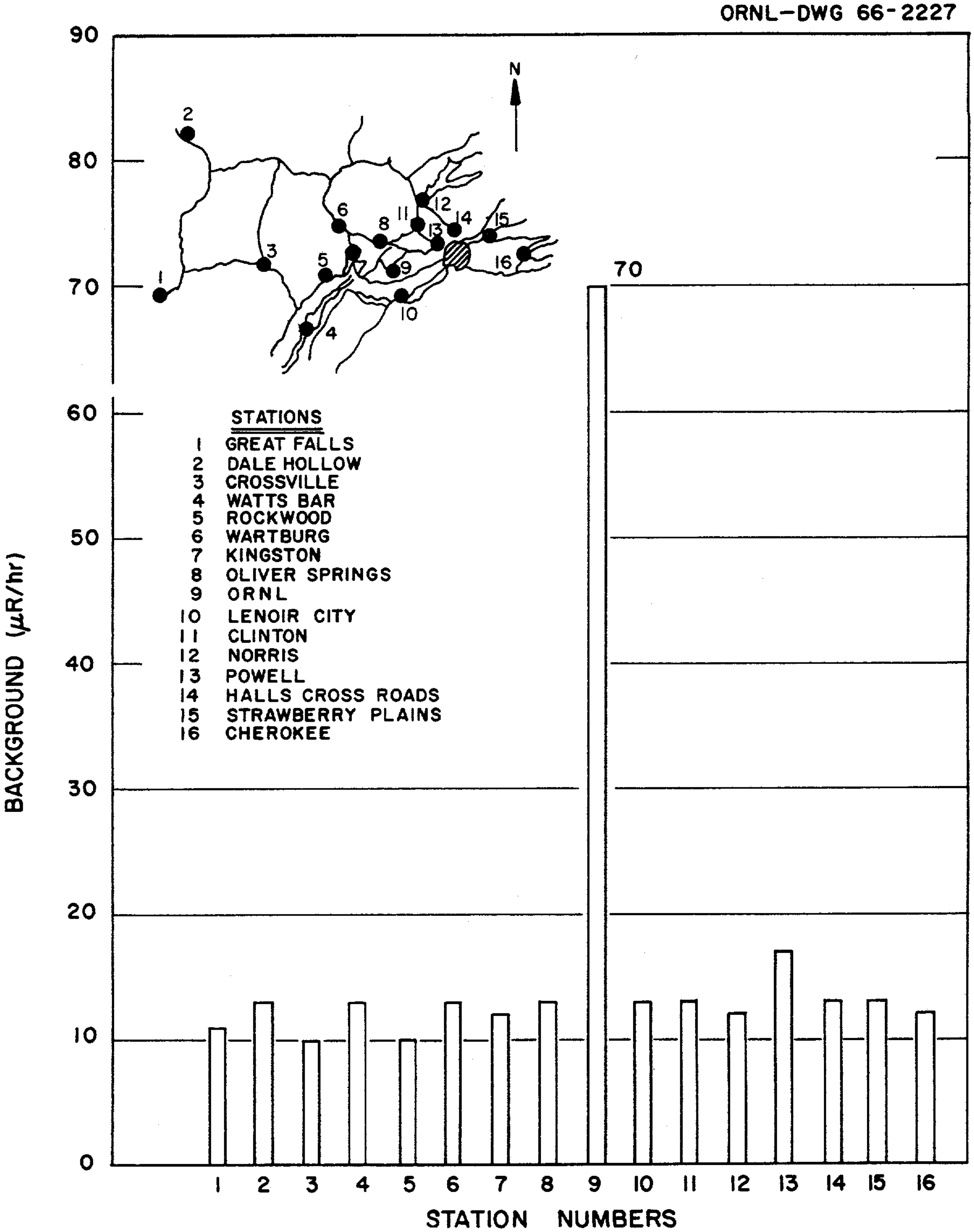


Fig. 15. Radiation Measurements Taken During 1965, 3 ft Above the Ground Surface out to Distances of 75 Miles from ORNL.

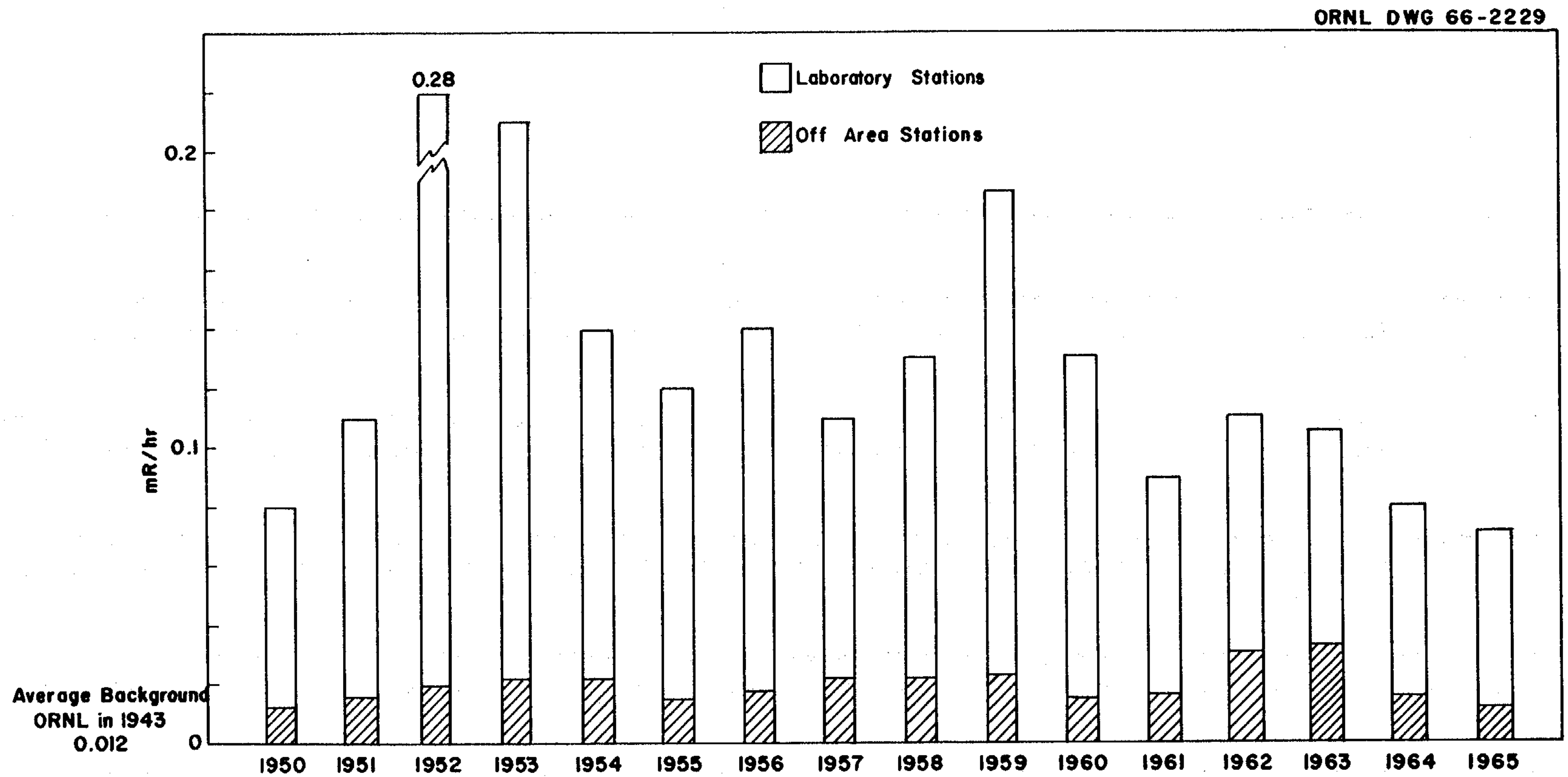


Fig. 16. Radiation Measurements Taken 3 ft Above the Ground Surfaces at ORNL Compared with Like Measurements Taken Elsewhere within the AEC Controlled Area for the Years 1950-1965.

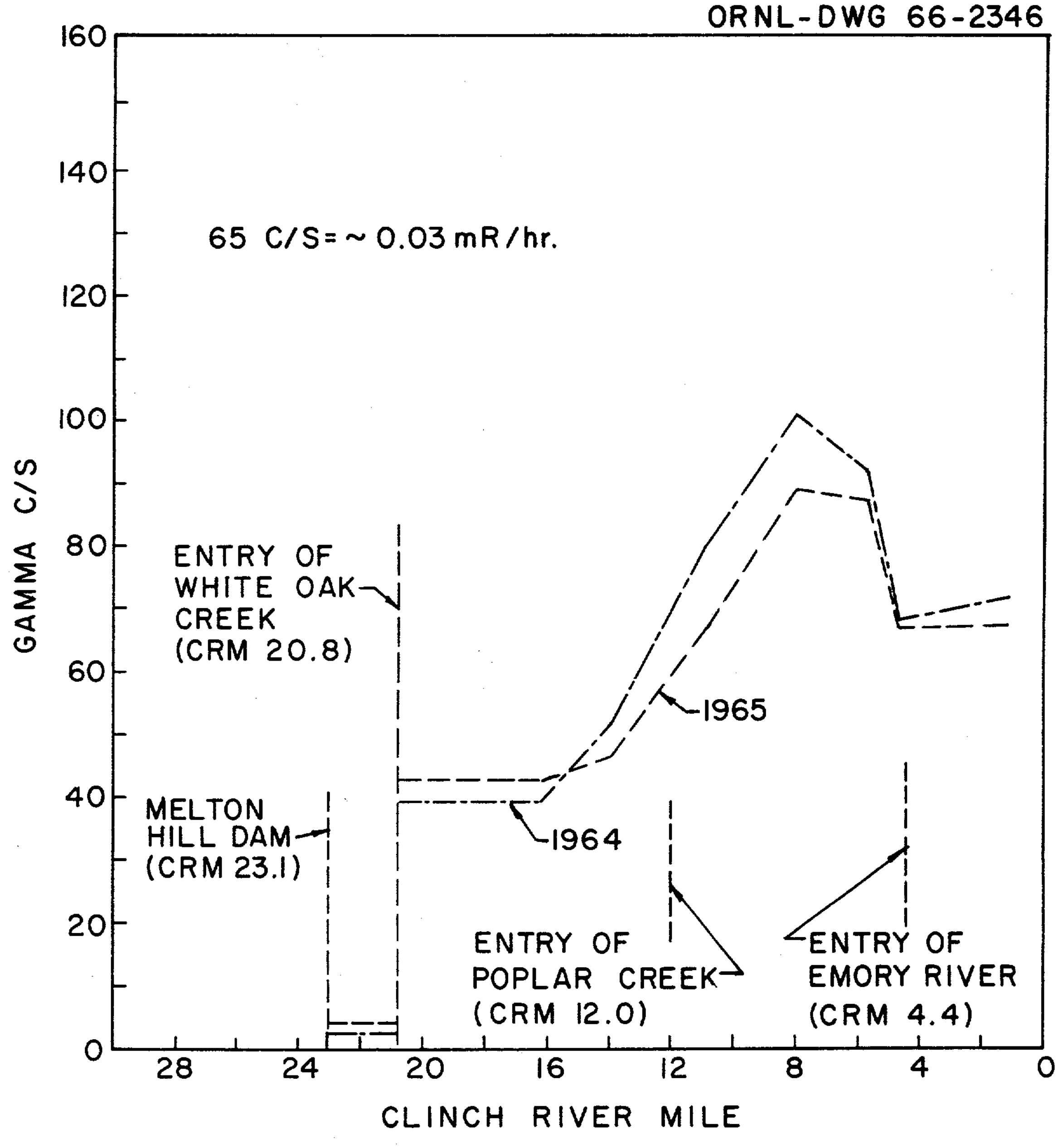


Fig. 17. Gamma Count at Surface of Clinch River Silt.

ORNL-DWG. 66-2345

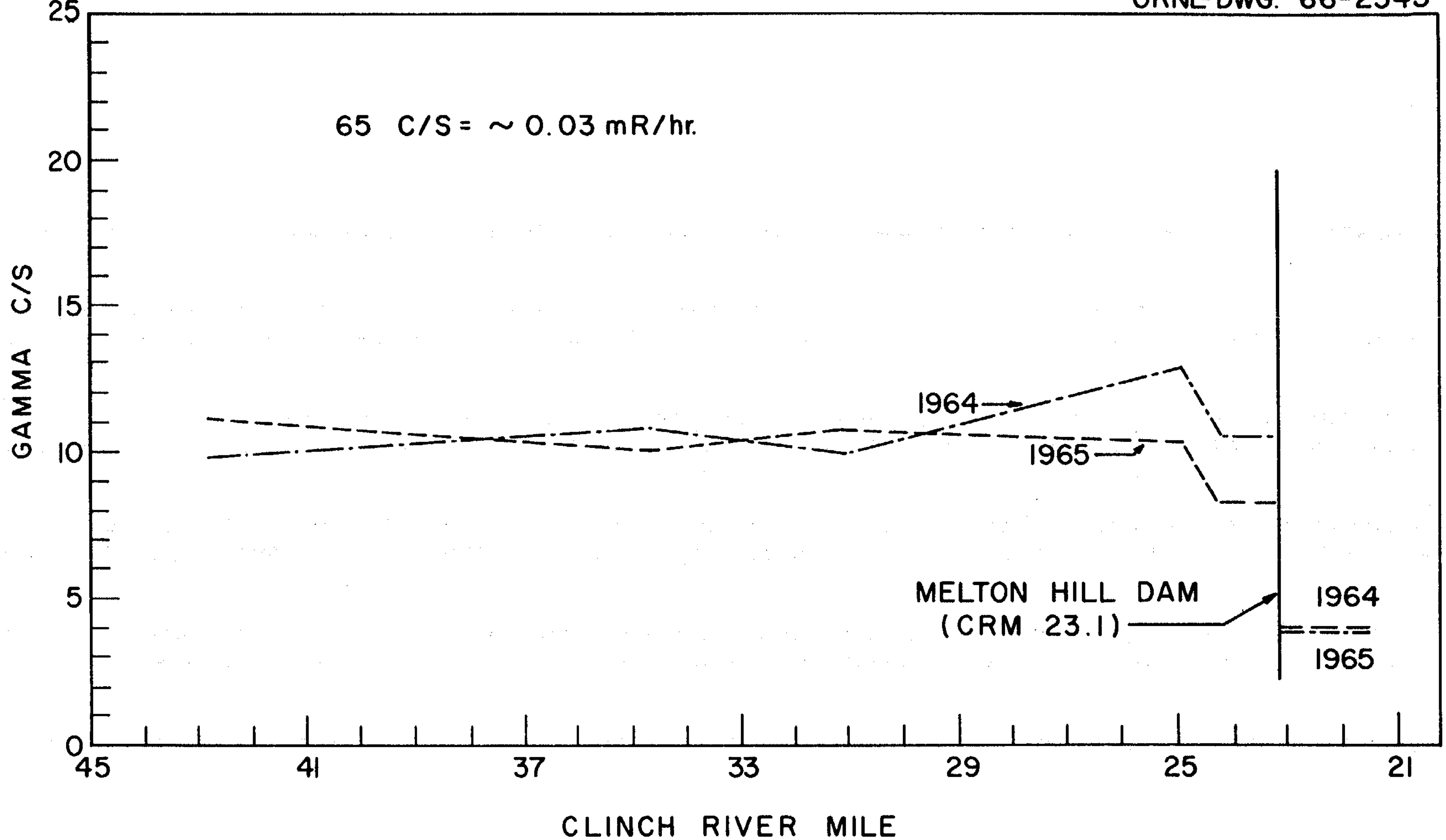


Fig. 18. Gamma Count at Surface of Clinch River Silt - Melton Hill Reservoir.

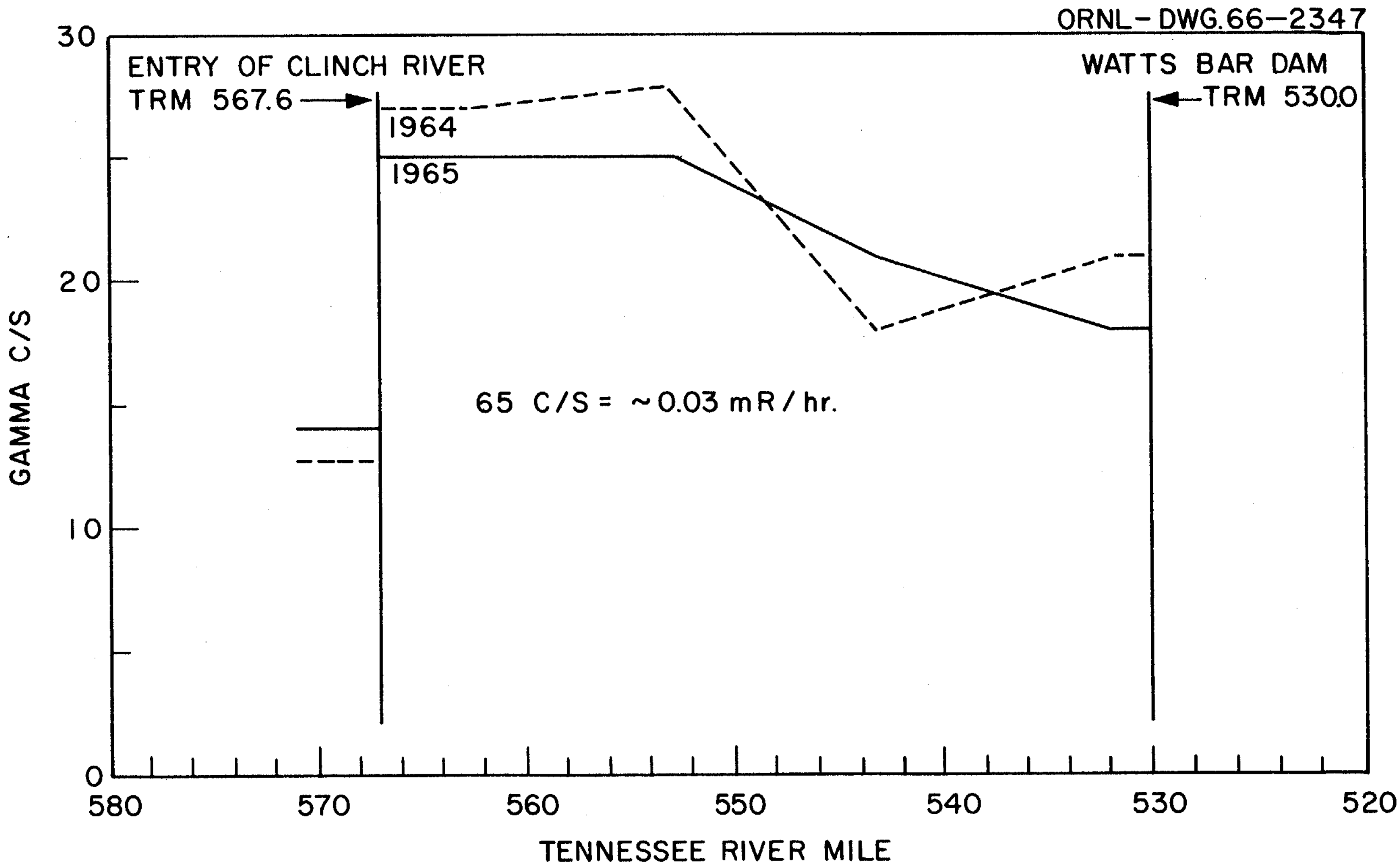


Fig. 19. Gamma Count at Surface of Tennessee River Silt.

Table 7 RADIONUCLIDES IN CLINCH AND TENNESSEE RIVER SILT - 1964-1965
(Units of 10^{-6} $\mu c/g$ of Dried Silt)

Location	^{137}Cs		^{144}Ce		^{90}Sr		^{60}Co		$^{103-106}Ru$		$^{95}Zr + ^{95}Nb$		TRE[a] + ^{90}Y (as ^{90}Y)	
	1964	1965	1964	1965	1964	1965	1964	1965	1964	1965	1964	1965	1964	1965
CRM 42.8	8.7	3.4	21	2.0	0.45	0.29	*	0.17	16	6.0	*	0.39	9.0	6.0
34.7	7.5	5.4	19	3.4	0.72	0.25	*	0.17	16	8.1	*	0.25	7.2	9.5
31.1	6.7	5.8	18	3.4	0.50	0.41	*	*	13	10	*	0.26	5.0	12
24.9	5.1	5.0	14	2.6	0.45	0.23	*	*	9.9	5.6	*	0.08	4.5	8.3
Average	7.0	4.9	18	2.9	0.53	0.30		0.08	13.7	7.5		0.25	6.4	8.9
CRM 21.5	0.68	0.62	1.4	0.57	0.29	0.38	0.36	0.23	0.99	0.91	*	0.15	*	*
16.3	172	145	5.6	2.6	1.4	1.2	11	10	12	2.1	*	*	6.0	7.9
14.0	62	49	6.8	1.2	0.50	0.41	6.2	6.4	20	11	*	0.10	3.2	3.1
11.0	77	58	13	1.7	0.77	0.92	8.9	8.4	34	9.9	*	0.10	4.3	6.2
8.0	105	95	14	2.1	0.88	0.70	12	12	43	22	*	0.16	7.9	12
5.8	89	114	14	2.8	0.83	0.81	11	15	40	28	*	*	6.4	11
4.7	100	96	15	3.1	0.86	0.79	11	14	41	25	*	0.26	8.2	9.9
1.1	67	78	21	2.8	0.59	0.79	7.8	13	37	23	*	0.20	6.8	11
Average	84	79	11	2.1	0.77	0.74	8.5	9.9	28	15		0.12	5.4	7.8
TRM 570.8	2.8	3.1	9.9	2.2	0.23	0.56	*	*	6.3	4.8	0.18	0.20	2.1	5.5
562.7	19	20	15	2.2	0.32	0.32	2.1	2.6	15	9.2	0.36	0.25	2.2	6.5
552.7	26	23	9.8	2.0	0.88	1.1	3.7	2.8	19	8.3	*	0.22	6.6	6.6
543.8	20	16	9.6	1.9	0.54	0.50	3.2	2.4	17	8.0	*	0.18	6.6	4.8
532.0	16	12	7.7	1.5	0.74	0.72	2.9	1.9	15	6.2	*	0.17	4.8	4.3
Average	17	15	10	2.0	0.54	0.65	2.4	1.9	14	7.3	0.11	0.20	4.5	5.5
Fort Loudoun Background Data														
TRM 615.8	--	2.8	--	1.6	--	0.23	--	0.03	--	4.2	--	0.19	--	3.9
604.4	3.3	3.1	8.1	1.5	0.47	0.34	*	*	5.8	5.0	0.59	0.14	2.4	4.8
Average		2.9		1.5		0.29		0.015		4.6		0.17		4.4

[a]TRE - total rare earths minus cerium
*None detected
--No samples taken

6.0 PERSONNEL MONITORING

It is the policy of Oak Ridge National Laboratory to monitor the radiation exposure of all persons who enter Laboratory areas where there is a likelihood of radiation exposure. Dose analysis is accomplished mainly through the use of personnel meters, bio-assays, and in vivo counting (whole body counter) techniques.

6.1 Dose Analysis Summary, 1965

6.1.1 External Exposures - No employee received a whole body radiation dose which exceeded the maximum permissible levels recommended by the Federal Radiation Council (FRC). The highest whole body dose received by an employee was about 4.4 rem or 37 percent of the maximum permissible annual dose. The range of doses for persons using ORNL badge-meters is shown in Table 8.

As of December 31, 1965, no employee had a cumulative whole body dose which exceeded the recommended maximum permissible dose as based on the age proration formula 5(N - 18) (Table 9). Only one employee had an average annual exposure rate that exceeded 5 rem per year of employment (Table 10). The average annual dose for this employee was 5.1 rem which was accrued over a period of about 14 years.

The highest cumulative dose to the skin of the whole body (head) received by an employee during 1965 was about 8.1 rem or 27 percent of the maximum permissible annual skin dose of 30 rem.

As of December 31, 1965, the highest cumulative dose of whole body radiation received by an employee was approximately 87 rem. This dose was accrued over an employment period of about 21 years and represented an annual exposure of about 4.0 rem.

The highest cumulative hand exposure recorded during 1965 was about 51 rem or 68 percent of the recommended maximum permissible annual dose to the extremities.

During 1965, except with the foreknowledge of Health Physics, no visitor-type meter was found to have sustained a positive (50 mrem) radiation dose.

6.1.1.1 External Dose - The average of the ten highest whole body doses of ORNL employees for each of the years 1959 through 1965 are shown in Figure 20. A downward trend in this value is apparent. The highest dose for each of those years is shown also.

The dose ranges versus the number of employees for each range for the years 1959 through 1965 are shown in Figure 21. Although the total number of employees increased slightly during the six-year period, the number of persons in the higher dose ranges decreased progressively.

The average annual dose to ORNL employees for the years 1959 through 1965 is the subject of Figure 22. This rather arbitrary quantity is obtained by dividing the sum of all doses for the year by the number of employees involved.

6.1.2 Internal Exposures - During 1965 there were no cases of internal exposure where the deposition of radioactive materials within the body was estimated to have averaged greater than one-half a maximum permissible body burden.[8]

Three employees continued to have estimated body burdens of transuranic alpha emitters (mainly ^{239}Pu) of 35 to 40 percent of the recommended maximum permissible value.[9] Health Physics procedures require that individuals who exceed 30 percent of a maximum permissible body burden be placed on a work assignment where the potential for internal exposure is reduced.

6.2 External Dose Techniques

6.2.1 Film Meters - Film meters are issued to all persons who have access to ORNL facilities in which there is a potential for radiation exposures in excess of Radiation Protection Guide (RPG) levels. Either an ORNL badge-meter (Figure 23) or a temporary pass-meter (Figure 24) may be used. Badge-meters are assigned to all ORNL employees, and to certain other persons who are authorized to enter ORNL facilities. Temporary pass-meters may be issued in lieu of badge-meters for short-term use.

NTA (nuclear track) film packets are included in all film meters. The NTA films are processed routinely if the badge-meter is assigned to an individual who normally works where there may be exposure to neutrons; otherwise the films would be processed only in the event of a nuclear accident.

Beta-gamma sensitive films from badge-meters issued to full-time employees and assignees are processed routinely each calendar quarter (or more frequently if necessary). Films used in other meters are processed as conditions of use may require. Films from meters issued to visitors are processed if there is a likelihood that a radiation exposure was incurred.

High-level radiation dosimetry components of the badge-meters (sulfur, gold, indium, and metaphosphate glass) are for use in the event that doses exceed the capability of the monitoring films.

For each ORNL division which had one or more employees who sustained a dose greater than 1 rem for the year, the number of employees so exposed are displayed in Figure 25. It may be noted that only 11 (of 29)

---

[7]Handbook 69 values are the basis for these determinations.

[8]AEC Manual Chapter 0502 requires an evaluation of the radiation exposure status of an employee when monitoring techniques indicate that a body burden equals or exceeds 50 percent of a maximum permissible limit.

divisions had employees with doses greater than 1 rem, only nine had employees with doses greater than 2 rem, and only three had employees with doses greater than 3 rem.

6.2.2 Pocket Meters - Pocket meters (indirect reading, ionization chambers) are made available at all principal points of entry to ORNL premises. A pair of pocket meters is carried for the duration of a work shift by persons who work in an area where the potential for an exposure of 20 mR or more exists during the work shift. Pocket meter pairs are processed each day by health physics technicians and readings of 20 mR or more are reported daily to supervision.

Pocket meters are used for a day-to-day record of integrated exposures and warn if excessive exposures occur.

Figure 26 is a display of the comparison between whole body doses as determined from film meters and the total recorded pocket meter readings for the ten highest whole body dose cases for the year 1965.

6.2.3 Hand Exposure Meters - Hand exposure meters (Figure 27) are film-loaded finger rings used to measure hand exposure. Hand exposure meters are issued on a weekly basis to persons for use during operations where it is likely that the hand dose is such as to exceed 1 rem during the week. They are issued and collected by Radiation Survey Unit personnel who determine the need for this type of monitoring and arrange for a processing schedule.

6.2.4 Metering Resume - Shown in Table 11 are the quantities of personnel metering devices used and processed during 1965. The number of films processed is less than the number issued, because those which are issued for only accident dosimetry are not processed unless there was a likelihood of exposure.

6.3 Internal Dose Techniques

6.3.1 Bio-Assays - Urine and fecal samples are analyzed for the purpose of making internal dose determinations. The frequency of sampling and the type of radiochemical analysis performed is based upon each specific radioisotope and the exposure potential. Because of the small quantities of radioactive material in most samples, qualitative analyses are not feasible, and only quantitative analyses for predetermined isotopes are performed routinely.

In most cases bio-assay data require interpretation to determine the dose to the person; computer programs are used for evaluation of extensive data on urinary excretion of ^{239}Pu. An estimate of dose is made for all cases in which it appears that one-third of a body burden, averaged over a calendar year, may be exceeded.

6.3.2 Whole Body Counter - The whole body counter (an in vivo gamma spectrometer) may be used for determining internally deposited quantities of most of the gamma ray-emitting substances, and many of the more energetic beta-emitting substances. Thus, it provides a direct method of determining body burdens of those substances.

6.4 Records and Reports

Most records and reports are prepared by electro-data processing (EDP) techniques through the use of high-speed digital computer systems. The IBM 7090, located at the Central Data Processing Facility (CDPF), turns out routine weekly, quarterly, and annual reports involving external dose data. (A typical weekly report is shown in Figure 28; a typical quarterly report is shown in Figure 29.) An IBM 1401, located at the Y-12 Plant, is used to provide the weekly pocket meter report (see Figure 30). Quarterly bio-assay listings are prepared by the IBM 7090 at CDPF; a weekly Bio-Assay Sample Status Report (Figure 31) is processed by the ORNL Math Panel utilizing a CDC 1604.

A monthly report based on preliminary results of analysis by the whole body counter (IVGS) is prepared by the IBM 7090 at CDPF.

Body burden estimates of ^{239}Pu are prepared in report form (usually quarterly) by use of the IBM 7090 at CDPF.

Permanent files are maintained at Health Physics and Safety Headquarters for each individual who is assigned an ORNL photo-badge-meter. An IBM card cross-indexing system is maintained at the principal monitoring stations for the purpose of expediting meter assignments. These IBM cards are compatible with the various computer programs and provide for the internal audit of all personnel monitoring record data.

Copies of the EDP reports, both temporary and final, are maintained for both the internal and external dose programs. Data used in the EDP program are stored on computer quality magnetic tapes. Data pertinent to the work of the dosimetry groups and information used in the non-EDP reports are maintained in record form by Dose Data personnel.

6.5 Program Developments

During 1965 the computer program for pocket meter readings was modified slightly to provide each week the cumulative frequency of meter usage by each employee during the quarter.

Minor modifications were made to the weekly and quarterly computer programs for bio-assay samples to obtain slight saving in computer usage time and costs and a decrease in time of report preparation and distribution.

Computer programs for auditing personnel exposure histories were developed during the year. These programs provide for an extensive examination of individual radiation dose records and monitoring information accumulated in record files and computer tapes.

A procedure was devised and effected for selecting and expeditiously analyzing, and reporting the doses recorded by, films from badges worn by persons who are most likely to have sustained radiation doses in excess of 300 mrem during a quarter. These films are selected on the basis of pocket meter total readings for the quarter.

The dosimetry components of the Temporary Pass-Meter, which may be issued for short-term usage for emergency metering, were revised to include a beta-gamma film packet, a neutron film packet, an indium foil, and a metaphosphate glass microdosimeter; all of which are contained in a plastic pouch. These items are reused throughout their useful life, and are processed only as required.

Table 8 DOSE DATA SUMMARY FOR LABORATORY POPULATION INVOLVING EXPOSURE TO WHOLE BODY RADIATION - 1965

Group	Number of Rem Doses in Each Range							Total
	0-1	1-2	2-3	3-4	4-5	5-6	6 up	
ORNL Employees	5476	124	38	9	1	0	0	5648
ORNL-Badged Non-Employees	540	1	0	0	0	0	0	541
TOTAL	6016	125	38	9	1	0	0	6189

Table 9 AVERAGE REM PER YEAR SINCE AGE 18 - 1965

	Number of Doses in Each Range				Total
	0-2.5	2.5-5.0	5.0-7.5	7.5 up	
ORNL Employees	5640	8	0	0	5648

Table 10 AVERAGE REM PER YEAR OF EMPLOYMENT AT ORNL - 1965

	Number of Doses in Each Range				Total
	0-2.5	2.5-5.0	5.0-7.5	7.5 up	
ORNL Employees	5621	26	1	0	5648

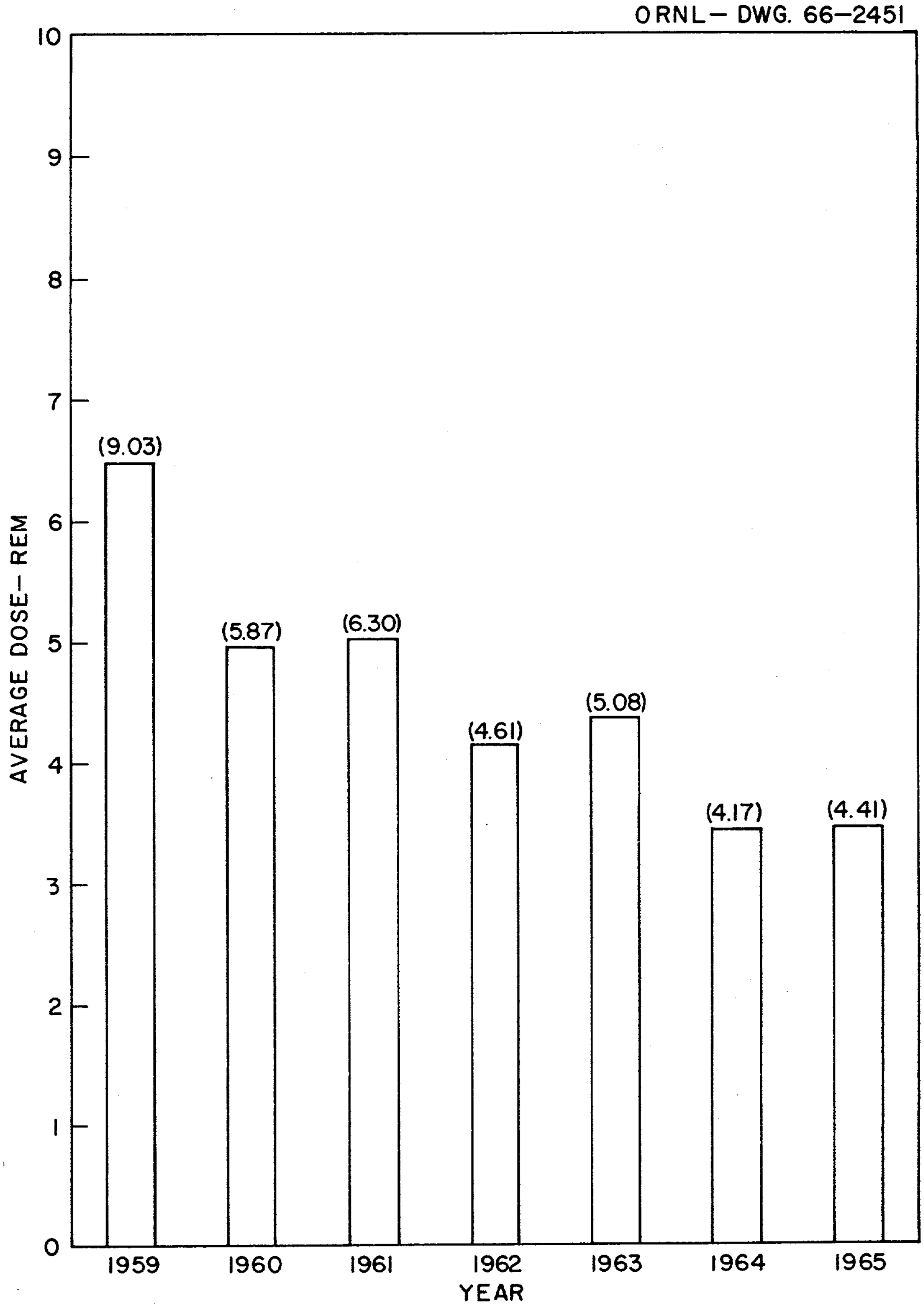


Fig. 20. Average of the Ten Highest Annual Whole Body Doses by Year (The Highest Individual Dose Shown in Parentheses).

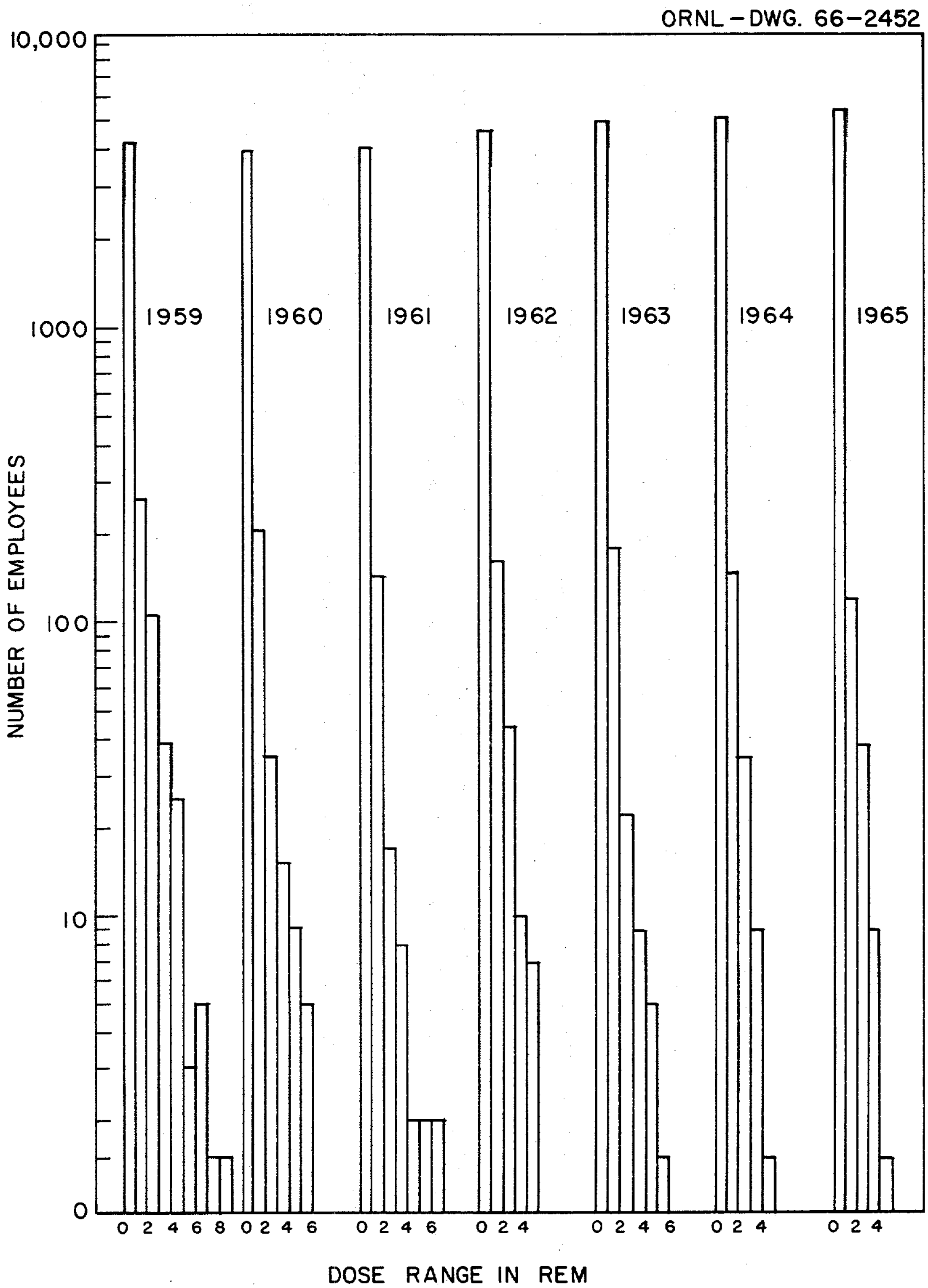


Fig. 21. Whole Body Radiation Dose Range for Employees - 1959-1965. (Does not include background radiation of about 100 mR/yr.)

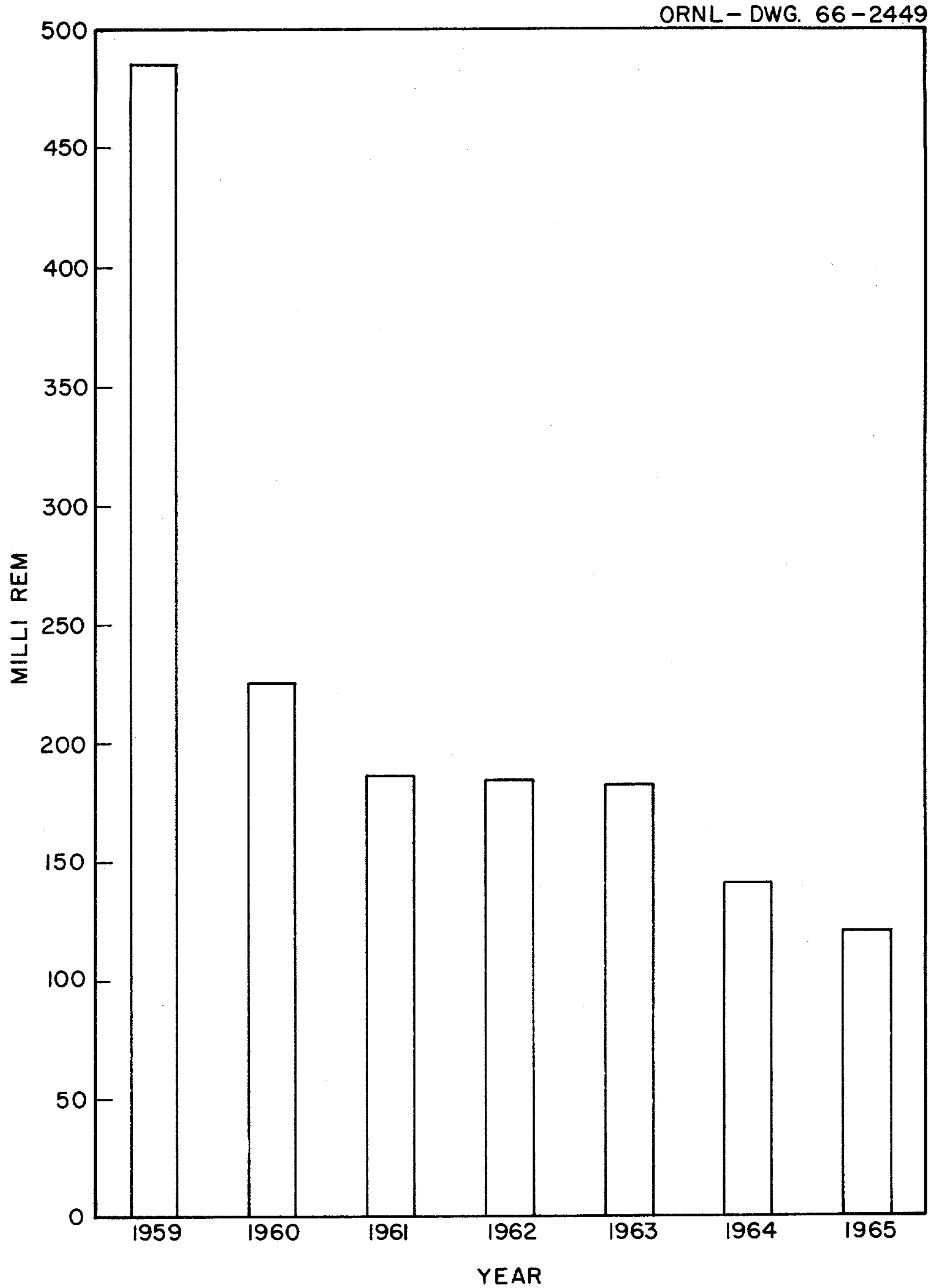


Fig. 22. Average Annual Whole Body Dose to the Average ORNL Employee. (Does not include background radiation of about 100 mR/yr.)

ORNL—LR—DWG 50112R

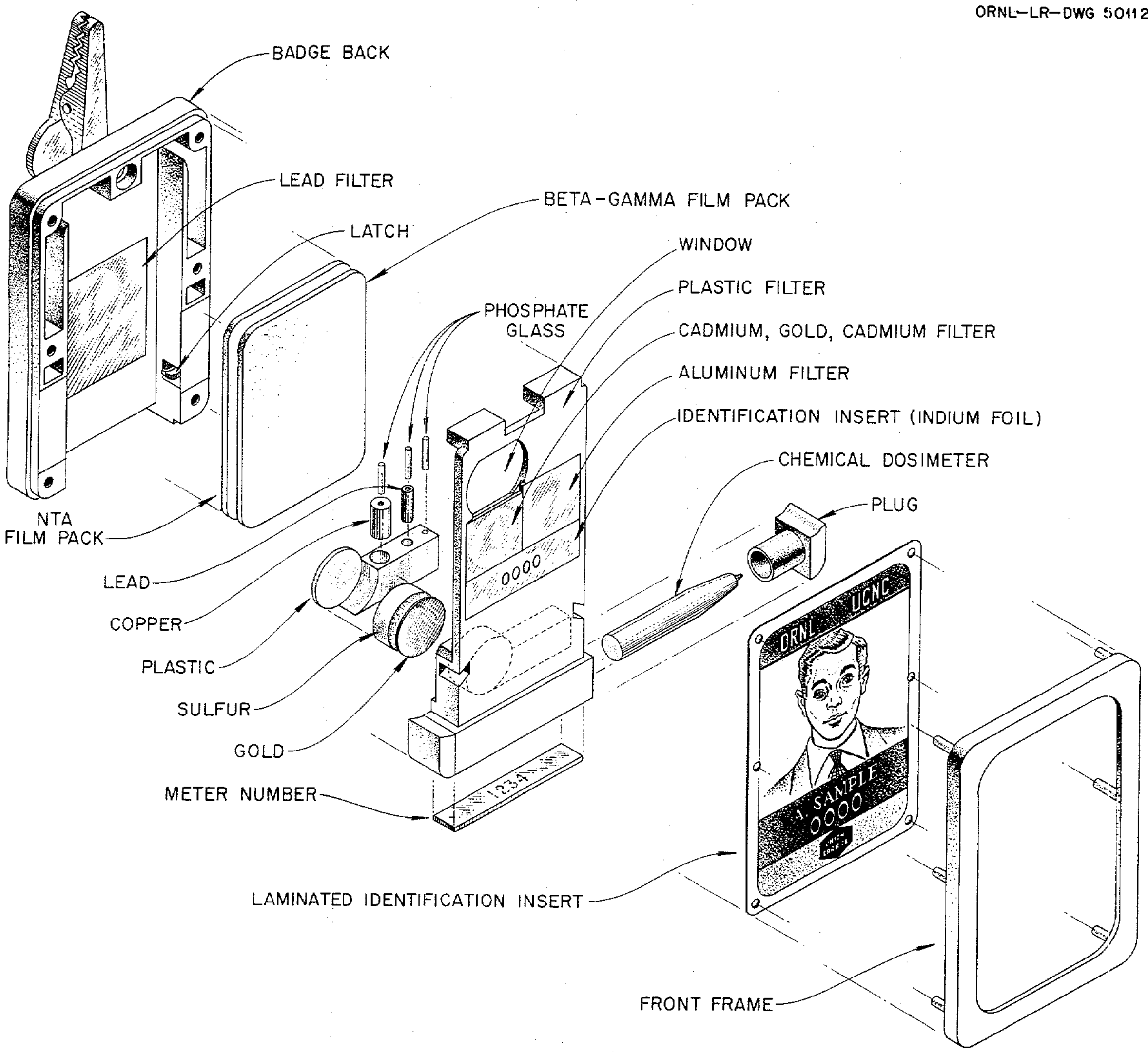


Fig. 23. ORNL Badge-Meter, Model II.

Fig. 24. Typical Temporary Security Passes Equipped with Monitoring Film.

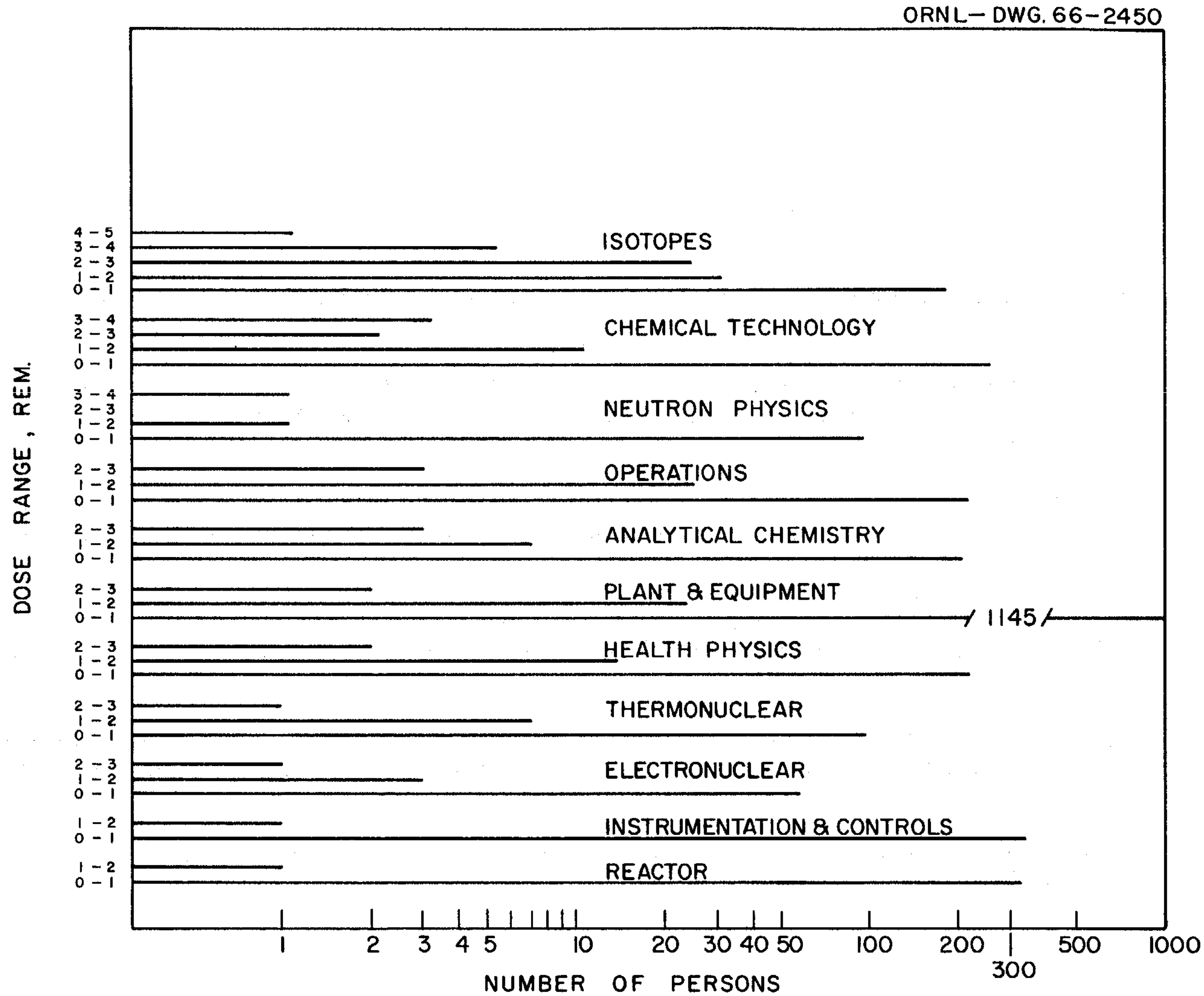


Fig. 25. Personnel Dose (Whole Body) by ORNL Division Having One or More Doses, One Rem or Greater, in 1965.

ORNL—DWG. 66-2448

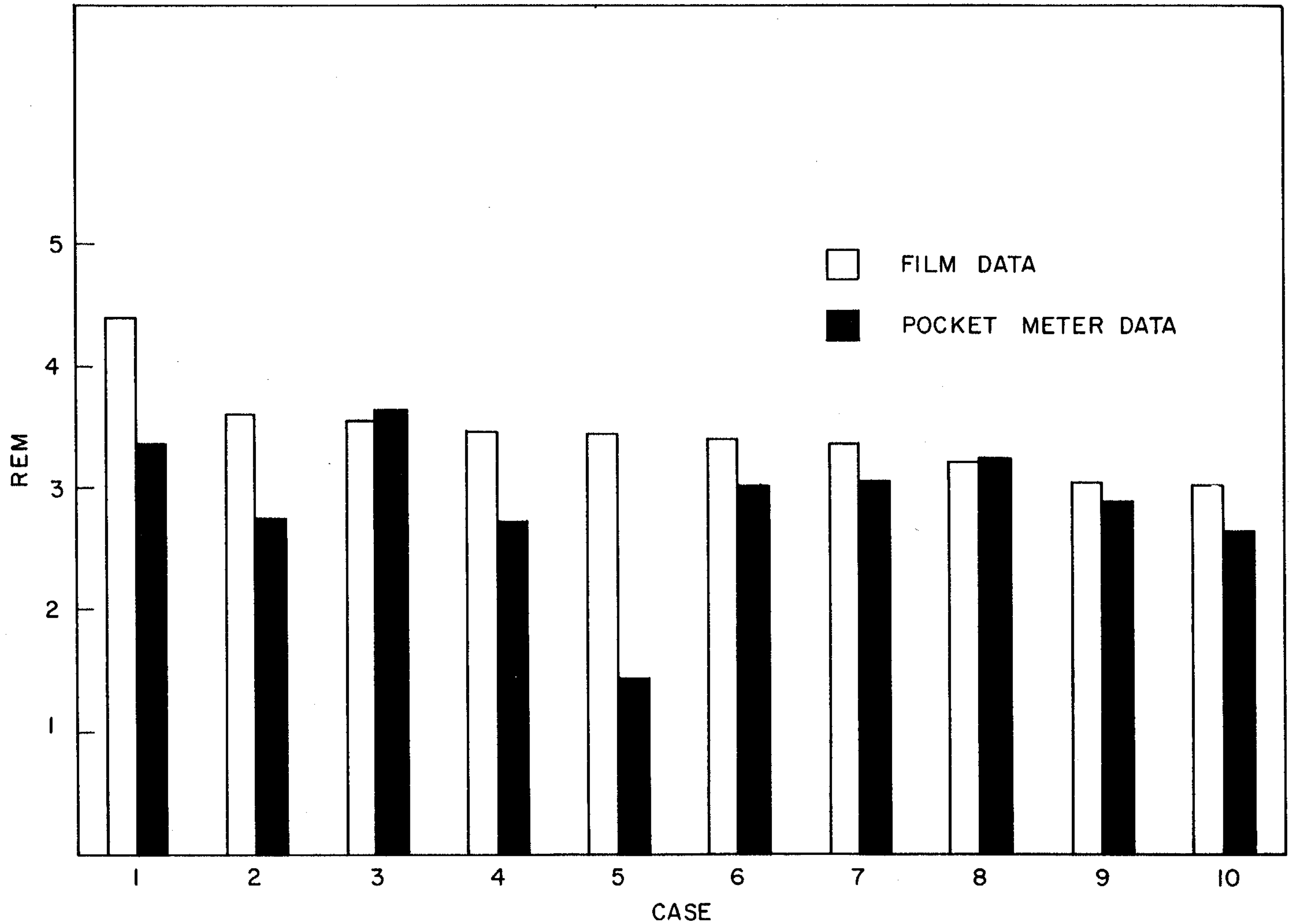


Fig. 26. The Ten Highest Whole Body Radiation Dose Cases Compared with Concurrent Pocket Meter Totals for 1965.

ORNL-DWG. 66-1814

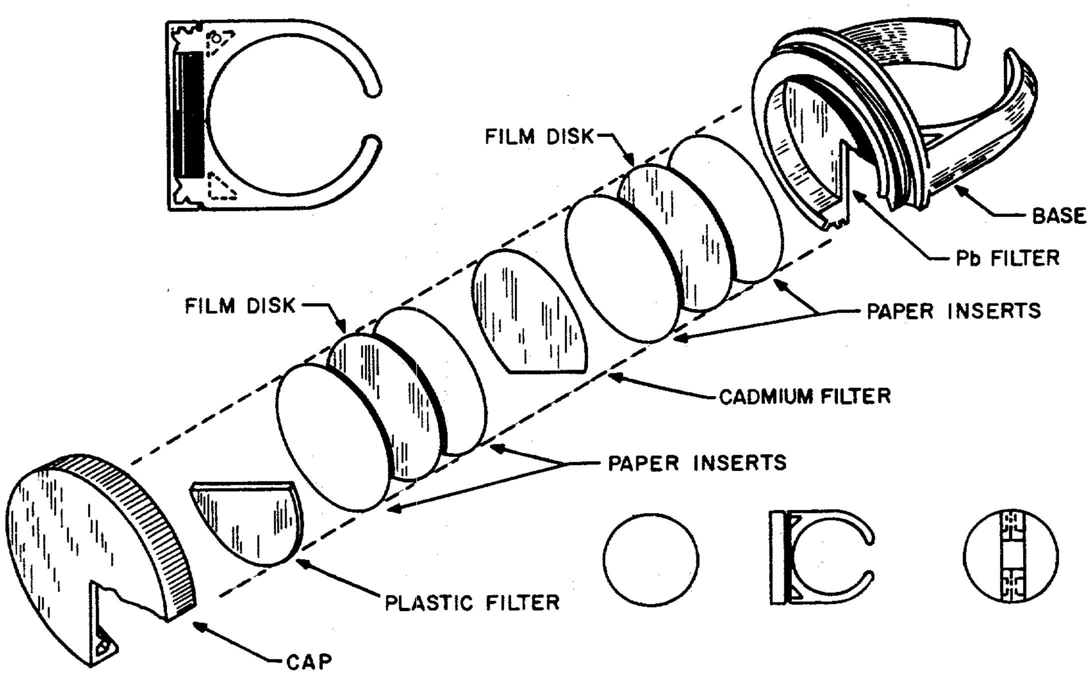


Fig. 27. Details of the ORNL Hand Exposure Meter.

Table 11 PERSONNEL METER SERVICES

	1964	1965
A. Pocket Meter Usage		
1. Number of Pairs Used		
ORNL	144,026	140,088
H. K. Ferguson	36,928	19,656
Total	180,954	159,744
2. Average Number of Users per Quarter		
ORNL	1,305	1,262
H. K. Ferguson	340	256
Total	1,645	1,518
B. Film Usage		
1. Films Used in Photo-Badge-Meters		
Beta-gamma	21,630	21,810
NTA	10,500	10,830
2. Films Used in Temporary Meters		
Beta-gamma	31,870	7,720
NTA	1,500	2,500
C. Films Processed for Monitoring Data		
1. Beta-Gamma	25,240	22,080
2. NTA	2,040	1,690
3. Hand Meter	1,740	1,610

ORNL DWG. 64-11676

Name	ID Number	Symbol	Dosimetry Dates		Meter Dose	
			Wk-Yr	Qtr-Yr	DS	DC
Last Name, Initials	PR. No.	PF	35-63	3-63	0.000	0.000
Last Name, Initials	PR. No.	PF	31-63	3-63	0.120	0.090
Last Name, Initials	PR. No.	PF	30-63	3-63	0.030	0.000
Last Name, Initials	PR. No.	PF	36-63	3-63	0.070	0.020
Last Name, Initials	PR. No.	PF	34-63	3-63	0.000	0.000
Last Name, Initials	PR. No.	PF	36-63	3-36	0.370	0.310
Last Name, Initials	PR. No.	PF	32-63	3-63	0.000	0.000
Last Name, Initials	PR. No.	PF	33-63	3-63	0.040	0.020
Last Name, Initials	PR. No.	PF	34-63	3-63	0.260	0.130
Last Name, Initials	PR. No.	PF	35-63	3-63	0.040	0.010

Fig. 28. Typical ORNL Film Monitoring Data.

ORNL DWG. 64-11675

HEALTH PHYSICS DIVISION
DEPARTMENT 3193
RADIATION SURVEY

Name	ID Number	Symbol	Date Wk-Yr	REM DS	REM DC	REM This Qtr DS	REM This Qtr DC	REM This Yr DS	REM This Yr DC	Total REM DC	A	DC/A
----	-----	PF	39-63	0.860	0.630	0.860	0.630	1.68	1.32	35.59	18	2.02
		PN	39-63	0.000	0.000							
----	-----	PF	39-63	0.340	0.240	0.340	0.240	0.34	0.24	0.24	1	0.80
----	-----	PF	39-63	0.020	0.010	0.020	0.010	0.02	0.01	5.21	14	0.38
----	-----	PF	39-63	0.070	0.040	0.070	0.040	0.30	0.19	18.38	16	1.19
----	-----	PF	39-63	0.390	0.310	0.390	0.310	1.40	1.14	2.74	20	0.14
----	-----	PF	39-63	0.350	0.150	0.350	0.150	0.77	0.49	9.60	17	0.56
----	-----	PEL	27-63	0.010	0.010	0.150	0.120	0.27	0.24	5.55	6	1.09
		PF	39-63	0.140	0.110							
		PN	39-63	0.000	0.000							
----	-----	PF	39-63	0.400	0.200	0.400	0.200	0.73	0.45	7.43	12	0.64
----	-----	PF	39-63	0.180	0.150	0.180	0.150	0.60	0.49	8.43	7	1.34
----	-----	PF	39-63	0.330	0.110	0.360	0.140	0.81	0.34	3.00	13	0.24
		PN	39-63	0.030	0.030							
----	-----	PF	39-63	0.180	0.080	0.180	0.080	0.51	0.33	29.82	18	1.68
		PN	39-63	0.000	0.000							
----	-----	PF	39-63	0.320	0.270	0.320	0.270	1.14	0.98	22.76	13	1.76
		PN	39-63	0.000	0.000							
----	-----	PF	39-63	0.420	0.290	0.420	0.290	1.85	1.11	15.86	16	1.04
----	-----	PF	39-63	0.320	0.140	0.320	0.140	0.67	0.46	8.96	11	0.84
----	-----	PF	39-63	0.390	0.210	0.390	0.210	1.21	0.72	33.62	18	1.87

Fig. 29. Typical ORNL Personnel Radiation Exposure Record.

ORNL DWG. 66-4519

DEPT XXXX	HP WK 52												
NAME	PR NO	DC	S	M	T	W	T	F	S	WK	QTR	F	SMB
---------------	-----			0	5	10	0			15	270	62	
---------------	-----	660		10	25		120			155	1140	57	DWQ
---------------	-----										35	11	
---------------	-----										10	1	
---------------	-----			10	10	0	10			30	220	41	
---------------	-----			10						10	115	62	
---------------	-----										5	20	
---------------	-----		0	10	10		20			40	560	54	D Q
---------------	-----										195	22	
---------------	-----											1	
---------------	-----		125	0		0				125	260	60	DW
---------------	-----			30	0	5	0			35	415	61	D
						ENTRIES		+		D	W	Q	COUNT
						23				5	2	2	12

Fig. 30. Typical Pocket Meter Weekly Report.

ORNL DWG. 66-4520

RESULTS THIS REPORT 12-20-65

Div. Code	Name	PR NO	HP AREA Number	Type Analysis	Receipt Date	Type Sample	Sample Priority	d/m/Sample	d/m/24 hrs
HP	----------	-----	3550	GAO	12-16-65	U	3		0
HP	----------	-----	3019	PUO	12-12-65	U	3		0
HP	----------	-----	3019	PUO	12-16-65	U	3		0
HP	----------	-----	3019	SRO	12-12-65	U	3		0
Div. Total 4									

Fig. 31. Typical Weekly Bio-Assay Sample Status Report.

7.0 LABORATORY OPERATIONS MONITORING

Radiation incidents are classified according to a severity index system developed over the past several years.[9] The method serves to index unusual occurrences according to degree of severity and permits a system of analysis regarding Health Physics practices among Laboratory operations. This report summarizes the unusual occurrence frequency rate and discusses some of the problems encountered among Laboratory facilities.

7.1 Unusual Occurrences

During 1965 there were 41 unusual occurrences recorded which represents approximately 29 percent increase over the number reported for 1964 (see Table 12) and approximately 17 percent below the five-year average of 48 for the years 1961 through 1965. Thirteen of the occurrences recorded during 1965 involved development work with the transuranic isotopes ^{242}Cm, ^{244}Cm, which because of the high specific activity involved presented unique containment problems to be solved. Previous to the year 1965, because of the amounts and availability, these particular isotopes had created no significant contamination problems.

During 1965, about 60 percent of the 41 events were classified as significant. For the purpose of this report an event is designated as significant when it is such as to (1) exceed a recommended maximum permissible limit and/or (2) requires work stoppage in an operation while cleanup measures are instituted following a radioactive contaminant release. Although approximately two-thirds of the 41 events were classified as significant according to the above definition, only one event occurring during 1965 required inter-departmental assistance before normal operations were resumed. There was only one external technical over-exposure case reported, and in only a few cases were minor work restrictions imposed. The frequency rate of unusual occurrences among the Laboratory divisions involved (Table 13) are known to vary in relationship to the following:

(1) Quantity of radioactive material handled.
(2) Number of radiation workers involved.
(3) The radiation hazard potential associated with a particular operation or facility.

7.2 Radiation Surveys

During 1965 the Radiation Survey personnel assisted the operating groups in keeping the contamination, air concentration, and personnel exposure levels well below the maximum permissible established limits. Through discussions, seminars, safety meetings and informal discussions with supervision of the operating groups, they assisted in reducing or eliminating a number of problems associated with radiation protection at the Laboratory. The following is a brief description of a few of the problems and method of solution.

[9] See Applied Health Physics Annual Report for 1963, ORNL-3665, pp. 14-15.

7.2.1 Demolition and Removal of East Portion of Building 3550 and Adjoining Semi-Works Facility - The demolition and removal of the east portion of Building 3550 and the semi-works facility during the summer of 1965 presented some unique problems involving radiation contamination control. These structures having been in continuous use for work with radioactive material since 1943 represented potential contamination sources which were difficult to localize and assess. The close proximity of these structures to the Instrumentation and Control facility (Building 3500) and to normal pedestrian traffic dictated that very rigid control procedures be put in force. The first preventive control measure instituted was to prevent possible airborne contamination from entering the nearby Instrumentation and Control facility. This was accomplished by sealing air intakes to the facility and constant surveillance at building entrances for possible tracking of contamination. Strict control measures involving the sealing of cut pipe ends, ducts, bagging of equipment and wet down procedures were effective in preventing the spread of contamination.

7.2.2 Annual Survey of X-ray Equipment - The annual survey of X-ray producing devices at X-10 and the X-10 portion of the Y-12 operations was completed during July, 1965. The equipment operators and supervision gave full cooperation during the survey. A number of X-ray diffraction units that failed to comply with that part of the requirements for providing a visible signal to indicate when the X-ray source is energized have had this feature added to the equipment.

7.2.3 Improved Containment at the High Radiation Laboratory Analytical Facility (HRLAF) - Floor and other surface contamination frequently resulted from use of the access drawers in the east wall of the HRLAF cells. During the past year two specially designed glove boxes were installed over two of the drawers, practically eliminating them as sources of contamination in the cell operating area.

Sample carriers and solid waste material are removed from the HRLAF cells via the west dock. These items generally are grossly contaminated with radioactive material and frequently the dock area becomes contaminated. Occasionally radioactive particulates were detected on the pavement adjacent to the dock as no containment wall existed on the west side. During the past year the dock was sealed and solid metal doors installed to prevent wind and rain from sweeping contaminants outside the building.

7.2.4 Construction ^{233}U Storage Facility, Building 3019 - During the past year a ^{233}U solution storage facility was constructed in the pipe tunnel of Building 3019. During initial phases of construction personnel were required to wear two suits of clothing with respiratory protection due to high level alpha contamination. After isolation of the storage area by two concrete walls, decontamination and paint bonds were recommended. Surface and air contamination levels were reduced so that much of the remaining work was done in a single suit of clothing and without respiratory protection. Considerable savings in suiting and unsuiting time resulted and worker efficiency improved considerably.

7.2.5 Health Physics Assistance During Shale Fracturing Experiment - Radiation Survey personnel assisted the operating group in planning zoning procedures and other radiation control measures during experiments at the Shale Fracturing Pilot Plant. There were some contamination problems encountered during the injections of the wastes but decontamination was performed satisfactorily by the operating personnel without significant exposure problems.

7.2.6 Health Physics Coverage at Project Salt Vault - Radiation Survey personnel assisted at the Salt Vault Project in Lyons, Kansas during the placing of 14 irradiated ETR fuel assemblies (approximately 10^6 curies of fission products) in the salt mine. Although there were extremely high radiation levels (up to 700 R/hr) in certain areas during some phases of the operation, the use of remote operating equipment, continuous monitoring and strict observance of zoning procedures aided in keeping all personnel exposures below maximum permissible levels. The maximum exposure received by personnel during the operation was 190 mrem. There was no indication of airborne activity or surface contamination in the work areas, mainly because the cans containing the fuel assemblies were thoroughly decontaminated at Idaho Falls before they were transported to the mine.

7.2.7 Health Physics Assistance During Startup of the MSRE and HFIR Reactors - During 1965, Radiation Survey personnel assisted in the criticality experiments and initial low-power operational cycles of both the Molten Salt Reactor Experiment and the High Flux Isotope Reactor by providing the monitoring manpower and equipment necessary to ascertain the adequacy of shielding design and construction. In general, the findings indicated no gross inadequacies or insurmountable problems at either of the new facilities at power levels which had been achieved at the end of the year.

7.2.8 Installation of a Conditional Air Supply for Plastic Suits - During the summer months of 1965 the problem of heat stress was encountered by workmen changing manipulators in the curium cells in Building 3028E. The workmen were required to work near the ceiling in two pairs of coveralls and a plastic suit. Air was supplied to the individuals in the plastic suits by connections to the air manifold distributor located at the top of the cells. Due to the amount of exertion required plus the high temperature in the work area some workmen showed signs of heat prostration. A detailed study of the problem was made by interested personnel from the Health, Plant and Equipment, Isotopes, and Health Physics Divisions. As a result of this study some revisions were made in the protective clothing requirements, and a conditional air supply was installed to supply cooled air to the individuals in the plastic suits. No further heat stress problems have been reported since these measures were adopted.

7.2.9 WC Waste Tank Equipment Improvements - During the year continuing spread of ^{242}Cm alpha contamination in the area of WC-10 waste tank brought about a joint study of the liquid waste situation at this location by Operations, Engineering and Health Physics personnel. The following steps were agreed upon by those concerned as means of solving the basic problem of contamination spread:

1. Remove loose contamination from walls and floor, in as far as feasible.
2. Pour two inches of concrete on top of entire floor area.
3. Paint pit walls and floor.
4. Replace the header and valves, blanking off lines no longer being used.
5. Place a stainless steel cover over entire pit and seal same.
6. Install an off-gas line to the 3029 stack.

Step 3 was completed in December, 1965, and work on steps 4, 5 and 6 is scheduled for completion during the early part of 1966.

7.2.10 Processing Enriched Uranium - The processing of approximately 16 kg of ^{235}U at a rate of 300 gram batches per day in B-1 and B-5 laboratories, Building 4500N, presented potential contamination and criticality problems that had to be considered prior to the start of the operation. Careful planning was mandatory for this particular program because of the proximity of the operations to adjoining laboratories with their full complement of working personnel.

Criticality control measures that were imposed contained limitations on mass implemented by a system of ^{235}U inventory management. In essence this procedure involved checking out to the next laboratory or process step the material batch on which the designated work had been completed before being allowed to receive a new batch for processing work or handling. This inventory method was rigidly followed throughout with care to obtain dual signatures of responsible individuals for all exit shipments and receipts of material.

The program was completed after several months of effort without problems related to contamination or personnel exposure. It is believed that the careful planning and the rigid control that were established played a major role in the success of this program.

7.2.11 Addition of Protective Shielding and Use of "Beam Stoppers" at the Van de Graaff Accelerator, Building 5500 - A series of experiments involving D(D, N) ^{3}He reactions conducted at the 3 MV accelerator generated neutron dose rates above that considered desirable for continual occupancy thereby requiring the addition of shielding between the target and control rooms. Attentuation of the neutron flux by at least an order of magnitude was achieved.

Fail safe type electrically operated "beam stoppers" for use at the Van de Graaff accelerator were recommended where entry into target areas during operation of the accelerator is practiced. Manually operated devices are being used until this recommendation is implemented.

7.3 Laundry Monitoring

A total of 988,139 articles of wearing apparel was monitored at the laundry during 1965. This was an increase of about 22 percent over the number monitored in 1964. Less than 2 percent of the items monitored were found to be contaminated. (Articles that are potentially contaminated are washed prior to monitoring.)

Of the 374,842 khaki garments monitored during the year, only 200 were found contaminated. This is a decrease of about 30 percent from last year (1964).

Table 12 UNUSUAL OCCURRENCES SUMMARIZED FOR THE 5-YEAR PERIOD ENDING WITH 1965

	1961	1962	1963	1964	1965
Number of Unusual Occurrences Recorded...........	75	55	43	29	41
A. Number of incidents of minor consequence involving personnel exposure below MPE limits and requiring little or no clean-up effort..................................	34	25	11	14	11
B. Number of incidents involving personnel exposure above MPE limits and/or resulting in special cleanup effort as the result of contamination..............................	41	30	32	15	30
1. Personnel Exposures.....................	7	7	4	9	12
a. Nonreportable overexposures with minor work restrictions imposed......	6	7	3	9	11
b. Reportable overexposures with work restrictions imposed................	1	0	1	0	1
2. Contamination of Work Area...............	40	30	32	15	28
a. Contamination that could be handled by the regular work staff with no appreciable departmental program loss...............................	37	28	30	14	27
b. Required interdepartmental assistance with minor departmental program loss.......................	3	2	2	1	1
c. Resulted in halting or temporarily deterring parts of the Laboratory program............................	0	0	0	0	0

Table 13 UNUSUAL OCCURRENCE FREQUENCY RATE WITHIN THE DIVISIONS FOR THE 5-YEAR PERIOD ENDING WITH 1965

Division	No. of Unusual Occurrences					5-Year Total	Per Cent Lab. Total (5-Year Period)
	1961	1962	1963	1964	1965		
Analytical Chemistry	3	5	9	3	6	26	10.7
Biology	1	1	2		1	5	2.1
Chemical Technology	19	13	11	3	8	54	22.2
Chemistry	2					2	.8
Plant and Equipment	4	3	1	2	2	12	4.9
Inspection Engineering		1		1		2	.8
Electronuclear Research	7				1	8	3.2
Health Physics			1*	1	2	4	1.6
Instrumentation and Controls				1		1	.4
Isotopes	9	18	5	12	10	54	22.2
Metals and Ceramics	5	2	1			8	3.2
Neutron Physics	3	3	2			8	3.2
Operations	12	6	9*	3	8	38	15.6
Physics	1	2	3	3	2	11	4.6
Reactor	7					7	2.9
Reactor Chemistry	1				1	2	.8
Solid State		1				1	.4
Thermonuclear	1					1	.4
TOTALS	75	55	43	29	41	243	100.0

*Shared responsibility with another division for one unusual occurrence.

8.0 INDUSTRIAL SAFETY

8.1 Medical-Accident Reports - (Includes First Aid, Serious Injuries and Disabling Injuries)

There were 1745 medical-accident cases reported during 1965. The operating divisions reported 1410 and the research divisions reported 335. The Plant and Equipment Division, with about 25 percent of the employees at the Laboratory, reported 1192, or about 68 percent of the total number. The second highest number reported was the 81 reported by the Operations Division. The number of medical reports by division for the divisions reporting more than 25 is shown in Table 14, Column I.

8.2 Serious Injuries

There were 97 serious injuries (as defined in ORNL Standard Practice Procedure 49B) reported during 1965. A listing of the divisions experiencing more than one serious injury for the year is shown in Table 14, Column II.

In presenting statistics on accidents, it should be obvious that the number of injuries in a division is not necessarily directly related to the enthusiasm, or lack thereof, for an efficient safety program. Other factors must be considered. For example, the division with the most disabling injuries not only has the largest number of employees, but also the greatest potential for serious injuries, such as climbing ladders, operating lathes and saws, using welding equipment, etc. Also, many of the injuries reported by this same division actually occurred while the employee was performing work in the facilities of another division.

8.3 Disabling Injuries

The Laboratory experienced 17 disabling injuries during 1965. A listing of disabling injuries by division is included in Table 15. For comparison, Table 15 also includes a listing of divisions experiencing the total of 56 disabling injuries for the five-year period 1961-1965.

The following is a listing of divisions which did not experience a disabling injury during 1965 and which have worked more than 1,000,000 man-hours since the last disabling injury.

Division	Man-Hours Since Last Disabling Injury
Analytical Chemistry	4,381,000
Chemical Technology	2,932,000
Physics	2,285,000
Technical Information	1,480,000
Solid State	1,461,000
Metals and Ceramics	1,048,000

The frequency rates of disabling injuries, by division, for the period 1961-1965 and for 1965 are shown in Table 16. A comparison of disabling injuries in the research divisions and the operating divisions in relation to percent of Laboratory employees is shown in Table 17.

Although the frequency rate is considered to be the most important statistic in determining the success of a safety program, the severity rate is significant as this is an index to the number of days lost from accidents. Table 18 shows a comparison of days lost, frequency rate, severity rate, and disabling index, by years, for the period 1961 through 1965. The disabling injury index, D.I., is the frequency rate multiplied by the severity rate.

The disabling injuries by division and type accident are shown in Table 19. Falls were the cause of one-third of the disabling injuries.

Table 14 A Comparison, by Divisions, of Medical Treatment Cases (MT), Serious Injuries (SI), and Disabling Injuries (DI)

Divisions with More than 25 MT Cases for 1965		Divisions with More than One SI for 1965		Divisions with More than One DI for Period 1961-1965	
Division	Number	Division	Number	Division	Number
Plant & Equipment	1192	Plant & Equipment	63	Plant & Equipment	28
Operations	81	Operations	6	Operations	7
Metals & Ceramics	64	Metals & Ceramics	6	Metals & Ceramics	5
Health Physics	53	Health Physics	5	Health Physics	4
Instr. & Controls	53	Lab. Protection	3	Lab. Protection	4
Chem. Tech.	44	Chemistry	2	Chemistry	2
					50
Anal. Chemistry	34	Isotopes	2		
Lab. Protection	26	Chem. Tech.	2	All Others	6
	1,547				56
		Reactor	2		
			91		
All Others	199				
	1,746				
		All Others	6		
			97		

Table 15 The Number of Disabling Injuries, by Division, for Period 1961-1965 and for 1965

Division	1961-1965	1965
Plant & Equipment	28	7
Operations	7	3
Metals & Ceramics	5	0
Health Physics	4	0
Laboratory Protection	4	2
Chemistry	2	2
Inspection Engineering	1	1
Isotopes	1	1
Instrumentation and Controls	1	1
Personnel	1	0
Director's	1	0
Neutron Physics	1	0
Total	56	17

Table 16 Frequency Rate of Disabling Injuries, by Division, for Period 1961-1965 and for 1965

Division	1961-1965	1965
Laboratory Protection	4.08	11.42
Operations	3.46	6.69
Plant and Equipment	2.51	3.45
Inspection Engineering	2.47	2.47
Health Physics	2.26	0.00
Metals and Ceramics	2.03	0.00
Chemistry	2.00	9.46
Personnel	1.50	0.00
Director's	1.46	0.00
Neutron Physics	1.37	0.00
Isotopes	0.81	3.83
Instrumentation and Controls	0.37	1.70
All Divisions	1.50	2.21

Table 17 A Comparison of Disabling Injuries in the Research Divisions and the Operating Divisions in Relation to Percent of Laboratory Employees

	Percent of Employees*	Percent of Disabling Injuries 1965	Percent of Disabling Injuries 1961-1965
Research Divisions	54.6	17.6	23.4
Operating Divisions	45.4	82.4	76.6
Operating Divisions:			
Plant and Equipment (P & E)	23.6	41.2	49.8
Other Operating Divisions	21.8	41.2	26.8
	45.4	82.4	76.6

*Based on report for January 1, 1966.

Table 18 Comparison of Days Lost, Frequency Rate, Severity Rate, and Disabling Index, by Years, 1961-1965

	1961	1962	1963	1964	1965
Days Lost	3709	2592	1220	1107	800
Frequency	1.55	1.45	1.55	1.07	2.21
Severity	576	377	172	148	104
Disabling Index	0.89	0.55	0.27	0.16	0.24

Table 19 Disabling Injuries by Division and Type Accident

DIVISION	OCCUPATION	TYPE OF ACCIDENT	TIME OFF (DAYS)
Plant and Equipment	Electrician	Fall	157
Plant and Equipment	Electrician	Drill Caught	45
Plant and Equipment	Machinist	Slipped (No Fall)	124+
Plant and Equipment	Carpenter	Fall	24
Plant and Equipment	Rigger	Fall	113
Plant and Equipment	Leadburner	Burn (Hydrogen)	27
Plant and Equipment	Painter	Turned Ankle	37
Operations	Laundry Worker	Burn	12
Operations	Pile Operator	Turned Ankle	18
Operations	Pile Operator Helper	Pressure Vessel	6
Lab. Protection	Guard	Vehicle	67
Lab. Protection	Fire Lieutenant	Fall	23
Chemistry	Chemist	Broke Glass	3
Chemistry	Co-op Student	Explosion	5
Insp. Engineering	Inspector	Fall	225
Isotopes	Technician	Turned Ankle	16
Instr. and Controls	Technician	Fall	34

9.0 LABORATORY ASSAYS

Laboratory Assays Units provide laboratory support to the Health Physics Monitoring Sections. These services include (1) the analysis of body fluids and excreta (bio-assay) for the monitoring of personnel for internal radiation exposure, (2) the radiochemical analysis of environs samples, (3) counting services for the environs monitoring and radiation survey programs, (4) autoradiograph, and (5) whole body counting (in vivo gamma spectrometry).

9.1 Bio-Assay Analysis

The number and types of analyses performed by the Bio-Assay Unit during 1965 are given in Table 20. A total of 7779 analyses were performed which include 7305 analyses on samples submitted by donors and 474 analyses on standard and blank samples analyzed for control purposes. Approximately 93 percent of the samples were analyzed for either the alpha emitters or strontium. The total number of analyses on samples submitted during 1965 increased by about two percent from the number processed during 1964.

9.2 Counting Facility

A total of 267,251 samples were processed by the counting facility during 1965. A tabulation of the number and types of samples counted is presented in Table 21. This total represents about a 10 percent decrease in the number of samples processed as compared with the previous year.

9.3 Environs Monitoring Sample Analysis

Table 22 presents the number and type of environs samples analyzed and the type of analysis performed on each type of sample. A total of 10,760 environs samples were analyzed during 1965 as compared to 11,946 samples analyzed in 1964. Analysis of environs monitoring samples may range from a single determination to as many as ten determinations per sample depending upon the radionuclides present. The methods used by the various analytical groups are generally described in the ORNL Master Analytical Manual.

9.4 Autoradiography

A total of 2,082 films were processed during 1965 in support of radioparticulate studies conducted by the Environs Monitoring Units.[10]

9.5 Whole Body Counter[11]

During the calendar year 1965 the whole body counting program included 1222 counts on 1079 persons; 1020 or about 83.5 percent of the

---

[10]Methods described in ORNL-2601, "Radioactive Waste Management at Oak Ridge National Laboratory".

[11]The Whole Body Counter is operated by the Health Physics Technology Section.

counts showed normal human spectra. Of the above total, 46 were initial counts made on persons involved in possible contamination incidents, only 30 of these showed any indication of internal contamination. Fifty-eight counts were for the purpose of further investigation of positive counts; 55 of the follow-up counts indicated contamination still present. In addition to the human counts, 105 counts were made for routine or special calibration or standard counts and 181 counts were made for the purposes of developing and improving in vivo counting capabilities. Table 23 lists the number of cases and maximum amount of isotopes detected in the counts which showed other than normal ^{40}K and ^{137}Cs.

The amount of ^{137}Cs that is considered normal due to assimilation of nuclear weapons fallout (i.e., non-occupational) has been subject to change over the past four years. Beginning in 1961 with the resumption of nuclear weapons testing, the level of ^{137}Cs detected in "unexposed" humans increased as did the amount of ^{137}Cs reaching the food chain. By 1964 the average burden of ^{137}Cs found in ORNL employees not exposed at work was roughly 0.025 μc. During the year 1965 the amount of ^{137}Cs found in the general population of ORNL employees began to decrease and approach pre-testing levels. The average ^{137}Cs whole body burden found in 51 otherwise unexposed male employees counted during January and February was 0.017 μc, the average for 51 male employees counted during October and November was 0.013 μc and the average for 34 male employees counted during December was 0.011 μc.

Approximately 27 percent fewer persons were counted and 26 percent fewer counts were made this year than were made in 1964. This slight reduction in counting (the 1965 totals are still 16 percent above the 1963 totals) is due to the change in the method of selecting and scheduling candidates for whole body counts. As of the second week in May, the program was completed under which an initial or baseline count was made on essentially everyone in the Laboratory with a potential for future exposure. Beginning with the third week in May, the Health Physics and Safety Section assumed the responsibility for selecting and scheduling employees for whole body counts on the basis of a five-part priority system. As before, top priority is given to those persons suspected of having sustained an exposure; second priority is given to persons being recounted as a follow-up to the findings of previous counts; the third priority group includes persons who work directly with radioactive materials, they are to be counted roughly four times a year; the fourth priority group includes persons who work in the areas where radioactive materials are handled but do not work directly with the materials themselves, they are to be counted roughly twice a year; the last priority groups includes any newly hired or other perions requiring baseline counts before beginning work with radioactive materials.

In August, 1965 a computer program written for the quantitative analysis of the gamma-ray spectra from very low level liquid waste samples was tested for use in analyzing whole body count spectra. The Fortran 63 program, Alpha,[12] was written by E. Schonfeld of the Chemical Technology

---

[12] E. Schonfeld, Alpha - A Computer Program for the Determination of Radioisotopes by Least-Squares Resolution of the Gamma-Ray Spectra, ORNL-3810, July, 1965.

Division. It has proved to be directly applicable to analysis of whole body count spectra without any changes from its original format. A previously reported computer program[13] written especially for the analysis of whole body count data and in use since January, 1963 proved to be unsatisfactory because of its limited capabilities and expense of operation. Program Alpha has been quite satisfactory in both these respects and offers much greater flexibility than the old program.

The case of two persons[14] exposed to inhalation of $^{90}SrTiO_2$ in January, 1964 has been followed by monthly in vivo chest counts. As of December 15, 1965, the chest burdens for both persons involved were estimated to be 0.28 μc ^{90}Sr in equilibrium with the same amount of ^{90}Y. This amounts to approximately 73 percent of the maximum permissible lung burden for ^{90}Sr - ^{90}Y.

---

[13]B. R. Fish et al, Health Physics Division Annual Progress Report, ORNL-3492, June 30, 1963, p. 194.

[14]B. R. Fish et al, Health Physics Division Annual Progress Report, ORNL-3697, July 31, 1964, p. 223.

Table 20 BIO-ASSAYS ANALYSES - 1965

Analytical Procedure	Number of Analyses	
Urine:		
Trans Pu	502	
Sr	2,628	
U	1,929	
TRE (total rare earths)	-	
^{3}H	104	
^{137}Cs	146	
^{239}Pu	1,628	
^{106}Ru	-	
^{32}P	-	
Other	218	
Total		7,162
Fecal:		
Gross Alpha	73	
Sr	8	
U	26	
Others	36	
Total		143
Miscellaneous:		
Blood, sputum, breath		-
Standards and blanks		474
GRAND TOTAL		7,779

Table 21 COUNTING FACILITY RESUME - 1965

Types of Samples	Number of Samples			Unit Total	Weekly Average
	Alpha	Beta	Gamma		
Survey Area Samples					
Smears	106,143	109,947		216,090	4,155
Air Filters	20,726	20,717		41,443	798
Environs Monitoring					
Air Filters	2,380	2,380		4,760	92
Gummed Paper		1,332		1,332	25
Rain Water		1,548		1,548	30
White Oak Lake Effluent	231	1,089		1,320	25
Animal Thyroids			267	267	5
Milk			491	491	9
GRAND TOTAL	129,480	137,013	758	267,251	5,139

Table 22 ENVIRONMENTAL MONITORING SAMPLES - 1965

	Sample Type	Type of Analysis	Number Samples
1.	Monitoring network filters	Gross beta, autoradiogram	1,767
2.	Gummed paper fall-out trays	Gross beta, autoradiogram	1,635
3.	CAM filters	Gross beta, autoradiogram	4,416
4.	Rain water	Gross beta	772
5.	White Oak Dam effluent	Gross beta, radiochemical, gamma spectrometry	1,134
6.	Clinch River water	Gross beta, radiochemical, gamma spectrometry	23
7.	Raw milk	Radiochemical	468
8.	Pasture grass	Radiochemical, gamma spectrometry	232
9.	Potable water	Radiochemical, gamma spectrometry	25
10.	Silt composites	Radiochemical, gamma spectrometry	21
11.	Animal thyroids	Gamma spectrometry	267
		TOTAL	10,760

Table 23. Measurable Radioactivity Found in Routine Whole Body Monitoring Program

Calendar Year 1965

Isotope	Maximum Amount Detected (μc)	Percent MPBB	Number Counts	Number Persons
^{46}Sc	0.004	0.008	1	1
^{51}Cr	Trace	---	1	1
^{56}Co	0.015	Not Listed	3	1
^{57}Co	0.065	0.03	10	2
^{58}Co	Trace	---	2	2
^{60}Co	0.15	1.5	17	14
^{65}Zn	0.060	0.1	3	3
^{90}Sr	0.35	92.0[a]	29	4
^{95}Zr - ^{95}Nb	0.060	0.3	6	3
^{106}Ru - ^{106}Rh	0.030	1.0	5	4
^{131}I	1.1	160.0	15	9
^{137}Cs	0.130	0.4	9	9[b]
^{144}Ce - ^{144}Pr	0.070	1.4	9	7
$^{188,189,190}Ir$	0.4	1.1	2	2[c]
^{203}Hg	0.016	0.4	1	1[d]
^{226}Ra	0.0005 μgm	0.5	2	1[e]
^{235}U	Trace	---	1	1
^{242}Cm	0.095	190.0	19	4
^{244}Cm	Possible, but not positively identified		3	2
Activity Unidentified			16	10

[a] Based on a MPLB of 0.76 μc for ^{90}Sr + ^{90}Y. The maximum amount detected, 0.35 μc as ^{90}Sr only, was the February count on one of the two $^{90}SrTiO_2$ exposure cases which have been followed since January, 1964. The estimate for the December count this year was 0.28 μc ^{90}Sr (+ 0.28 μc ^{90}Y).

[b] Body burdens in excess of 0.045 μc ^{137}Cs are taken to indicate occupational exposure.

[c] The estimate of 1.1 percent is based on the MPBB for ^{190}Ir.

[d] In addition to the one employee mentioned in the table, two employees, recently hired from another radioisotope processing laboratory, were found to have 0.022 and 0.03 μc of ^{203}Hg. Two other employees were checked who had received injections of radioactive mercury as part of medical test procedures. The first of these was detected during a routine body count and found to have 0.17 μc ^{203}Hg; the length of time since injection was not known. The second employee was known to have been injected when counted. His body burden was estimated to be around 40 μc ^{203}Hg, approximately one month post-injection.

[e] One of three persons sent from another government agency installation for counting.

10.0 HEALTH PHYSICS INSTRUMENTATION

The Health Physics Division shares with the Instrumentation and Controls Division the responsibility for the development of electronic radiation monitoring instruments used in the Laboratory health physics program. Normally the Health Physics Division is responsible for determining the need for new instrument types and modifications to existing types, specifies the health physics requirements for design, and approves the final design. The Health Physics Division is also responsible for calibrating all instruments used in the health physics program and is allocated the funds for maintenance of these instruments. Maintenance is performed or crossordered by the Instrumentation and Controls Division.

Non-electronic personnel monitoring devices are designed, tested, calibrated, and maintained by Health Physics Division personnel.

10.1 Instrument Inventory

The electronic instruments used in the health physics program are divided, for convenience in servicing and calibrating, into two classes: the first class includes battery-powered portable instruments; the second class includes the stationary instruments that are A.C. powered. Portable instruments are assigned and issued to the Radiation Survey Units. Stationary instruments are the property of the Laboratory division which has the monitoring responsibility in the area in which the instrument is located. Table 24 lists portable instruments assigned at the end of 1965; Table 25 lists stationary instruments in use at the end of 1965. There were net increases in 1965 of 109 portable instruments and 108 stationary instruments.

During 1965, 246 new pocket meters and 309 new fiber dosimeters (200 mR range) were issued by ORNL Stores. Most of the meters issued were replacements for instruments which had been lost or damaged.

Inventory and Service Summaries for health physics instruments are prepared on an IBM 7090. These computer programmed reports enable the Instruments Group to maintain a current inventory on most health physics instrument requirements.

The allocation of stationary health physics monitoring instruments by divisions is shown in Table 26. Discrepancies between Table 26 and Table 25 are mainly because of differences in inclusion of "other" or miscellaneous instrument types and to what extent instruments in ORNL facilities at the Y-12 Plant are included.

10.2 Calibration Facility

The Health Physics Division maintains a calibration facility for the calibration and maintenance of portable radiation instruments and personnel metering devices. The facility is equipped with calibration sources, remote control devices, and shop space for the use of Instrumentation and Controls

Division maintenance personnel. Health Physics personnel assign, arrange for maintenance of, calibrate, provide delivery services for, and maintain inventory and servicing data of all portable health physics survey instruments.

Portable instruments are serviced (1) whenever repairs are needed, (2) at least once each two months for those which have replacement-type batteries, and (3) at least once each three months for those instruments which have "permanent" (rechargeable) batteries. The number of calibrations of portable instruments for FY 1965 is shown in Table 27.

Stationary instruments are calibrated by Calibrations Group personnel or by Radiation Survey personnel who use sources which are designed, standardized, and provided by the Calibrations Group.

10.3 Instrumentation Developments

Experiments were conducted to determine the response of field-type radiation survey instruments to beta radiation from small-area sources, in order to evaluate their capability for estimating personnel skin dose exposures. On the basis of the data, an instrument was developed for this purpose (see 2, below), and a report is being prepared.

The ORNL nuclear accident dosimetry system was evaluated in a series of tests conducted at the HPRR, in which the systems of several AEC facilities and private vendors were evaluated. Our results were in very good agreement with the known values.

A number of editorial changes were prepared and submitted early in 1965 for inclusion in the Health Physics Instrument Manual, ORNL-332. It is hoped that they will be available to Manual assignees before long.

During 1965, instrumentation development included the following:

1. The prototype air monitor for detecting alpha emitters in the presence of radon was received and tested. As anticipated, this device permits a factor of ten improvement over the standard alpha air monitor (Q-2340) in the detection of low levels of airborne, particulate associated, alpha particle emitters in the presence of normally occurring radon-thoron daughters and moderate levels of gamma radiation. The principle of operation is based upon a cancellation, in the output counting rate, of the counts produced by alpha particles from the emanation daughters.

2. In order to provide an instrument which may be used for measuring the dose rates to skin from small area sources of beta radiation, a portable count rate meter in combination with a specially designed end window GM counter was designed. A prototype of this instrument has been received and is undergoing tests and calibration with several types and distributions of beta sources. It is anticipated that acceptably accurate measurements of beta dose rates may be made in the field with this instrument.

3. Because an air proportional counter for alpha particle monitoring may have advantages in terms of weight and economy over other large-area

detectors, and because one of the research groups at ORNL wished to obtain a quantity of instruments (Physics International, Model PIC 1000) which are equipped with an air proportional counter, two such instruments were purchased and tested. The results of these tests are such that the instrument in its present design is unacceptable for Health Physics approval. However, the air proportional probe seems to have merit, and a portable, battery powered, counting rate meter is being designed by the ORNL Instrumentation and Controls Division for use with the probe.

4. The Model Q-1975, portable, scintillation alpha counter has been redesigned to incorporate a panel meter and three rate ranges, in addition to the counting register in the original design. The new instrument is designated Model Q-2789, and it will become the standard model for future procurement.

5. The Model Q-2004, thermal neutron survey meter was modified slightly and repackaged. The new design, Model Q-2824, is much smaller and weighs two pounds less than the older model, which it supersedes for future procurement.

6. An instrument, with detectors and read-out especially adapted for evaluating personnel contamination for medical purposes, has been designed, fabricated, tested and calibrated and installed in the ORNL Health Center (Figure 32).

Table 24 PORTABLE INSTRUMENT INVENTORY - 1965

Instrument Type	Instruments Assigned 1965	Instruments Retired 1965	Assigned Inventory Jan. 1, 1966
GM Survey Meter	27	5	415
Cutie Pie	65	13	449
Juno	0	1	33
Alpha Survey Meter	28	4	170
Neutron Survey Meter	11	0	55
Miscellaneous	1	0	38
TOTAL INVENTORY	132	23	1160

Table 25 INVENTORY OF STATIONARY, RADIATION MONITORING INSTRUMENTS FOR THE YEAR - 1965

Instrument Type	Installed During 1965	Retired During 1965	Total Jan. 1, 1966
Air Monitor, Alpha	18	1	79
Air Monitor, Beta	16	0	173
Air Monitor, Environmental	1	0	38
Hand-Foot Monitor	4	0	32
Lab Monitor, Alpha	20	0	106
Lab Monitor, Beta	17	1	165
Monitron	17	3	230
Other	20	0	128
TOTAL	113	5	951

Table 26 HEALTH PHYSICS FACILITY MONITORING INSTRUMENTS
DIVISIONAL ALLOCATION - 1965

ORNL Division	α Air Monitor	β Air Monitor	α Lab Monitor	β Lab Monitor	Monitron	Other	Total
Analytical Chemistry	4	12	8	15	14	5	58
Chemical Technology	31	34	36	20	31	27	179
Chemistry	2	6	9	13	19	6	55
Metals and Ceramics	9	16	13	17	11	12	78
Reactor	5	12		9	11	7	44
Isotopes	17	31	14	34	53	24	173
Operations	1	44	2	17	62	17	143
All Others	10	18	24	40	29	100	221
TOTAL	79	173	106	165	230	198	951

Table 27 CALIBRATIONS RESUME - 1965

		1964	1965
A.	Portable Instruments Calibrated		
	1. Beta-Gamma	3,923	4,065
	2. Neutron	115	142
	3. Alpha	943	985
	4. Pocket Chambers and Dosimeters	3,105	2,843
B.	Films Calibrated		
	1. Beta-Gamma	2,272	1,700
	2. Neutron	20	18

Fig. 32. Medical Monitor for Superficial Contamination.

11.0 PUBLICATIONS AND PAPERS

D. M. Davis, "Action Levels for Radiation Control at Oak Ridge National Laboratory", American Industrial Hygiene Association Journal, Vol. 26, March-April, 1965.

H. H. Abee and D. M. Davis, "Radiation Dose Received by Passengers and Crew on Planes Carrying Radioisotope Shipments", Tennessee Valley Industrial Health Conference, Gatlinburg, Tennessee.

Health Physics Manual, "Procedures and Practices for Radiation Protection", July 1, 1965.

12.0 VISITORS AND TRAINING GROUPS

During 1965, there were 146 visitors to Health Physics and Safety, as individuals or in groups, for training purposes. Table 28 is a listing of the training groups which consisted of four or more persons.

Table 28 TRAINING GROUPS IN HEALTH PHYSICS AND SAFETY FACILITIES DURING 1965

Facility	Number	Training Period
ORSORT, Students	32	1/13/65
TVA, EGCR	4	3/29/65 (6 wks)
Taft Rad. Health Center	35	4/13/65
ORINS (Public Health) (Univ. of Arkansas)	7	4/29/65
AEC-Fellowship	19	6/14 to 8/28/65
U. of North Carolina (Public Health)	7	9/7 to 9/11/65
Brookhaven Nat'l. Lab.	4	10/28/65
French AEC	10	11/8 and 11/9/65

ORNL-3969
UC-41 — Health and Safety

INTERNAL DISTRIBUTION

1. Biology Library
2-4. Central Research Library
5. Laboratory Shift Supervisor
6. Reactor Division Library
7-8. ORNL -- Y-12 Technical Library
Document Reference Section
9-43. Laboratory Records Department
44. Laboratory Records, ORNL R.C.
45. H. H. Abee
46. R. G. Affel
47. T. A. Arehart
48. S. I. Auerbach
49. J. A. Auxier
50. J. K. Bair
51. P. S. Baker
52. L. H. Barker
53. S. E. Beall
54. R. H. Beidel
55. C. G. Bell
56. M. Bender
57. S. R. Bernard
58. D. S. Billington
59-64. R. D. Birkhoff
65. N. E. Bolton
66. C. J. Borkowski
67. G. E. Boyd
68. J. W. Boyle
69. R. B. Briggs
70. F. R. Bruce
71. T. J. Burnett
72. G. C. Cain
73. A. D. Callihan
74. W. R. Casto
75. R. L. Clark
76. A. J. Cook
77. J. A. Cox
78. F. L. Culler
79-90. D. M. Davis
91. D. G. Doherty
92. L. G. Farrar
93. D. E. Ferguson
94. B. R. Fish
95. J. L. Fowler
96. J. H. Frye, Jr.
97. C. B. Fulmer
98. D. C. Gary
99. J. H. Gillette
100. W. Y. Gissel
101. W. R. Grimes
102. E. D. Gupton
103. C. C. Harris
104. J. C. Hart
105. C. E. Higgins
106. A. Hollaender
107. A. S. Householder
108. J. T. Howe
109. H. C. Hoy
110. T. W. Hungerford
111. W. H. Jordan
112. G. W. Keilholtz
113. C. P. Keim
114. M. T. Kelley
115. F. Kertesz
116. C. V. Ketron
117. E. M. King
118. C. E. Larson
119. T. A. Lincoln, M.D.
120. R. S. Livingston
121. H. G. MacPherson
122. F. C. Maienschien
123. J. D. McLendon
124. R. A. McNees
125. A. J. Miller
126. E. C. Miller
127-129. K. Z. Morgan
130. M. L. Nelson
131. A. R. Olsen
132. F. L. Parker
133. E. C. Parrish
134. M. E. Ramsey
135. M. L. Randolph
136. L. P. Riordan
137. J. B. Ruch
138. A. F. Rupp
139. G. S. Sadowski
140. H. M. Sartelle
141. N. B. Schultz
142. H. E. Seagren

143. M. J. Skinner
144. A. H. Snell
145. W. S. Snyder
146. W. M. Stanley
147. E. G. Struxness
148. E. H. Taylor
149-163. A. D. Warden
164. A. M. Weinberg
165. R. H. Winget
166. E. J. Witkowski
167. G. Young
168. J. E. Cantlon (consultant)
169. J. C. Frye (consultant)
170. J. B. Hursh (consultant)
171. J. L. Magee (consultant)
172. H. O. Wyckoff (consultant)

EXTERNAL DISTRIBUTION

173. C. P. Straub, Public Health Service, Robert A. Taft Sanitary Engineering Center, Cincinnati, Ohio
174. O. W. Kochtitzky, Tennessee Valley Authority, 717 Edney Building, Chattanooga, Tennessee
175. C. S. Shoup, Biology Division, USAEC, Oak Ridge, Tennessee
176. J. Thomas, Instructor of Public Health, Yale University
177. J. A. Swartout, 270 Park Avenue, New York 17, New York
178. Sakai Shimizu, Institute for Chemical Research, Kyoto University, Kyoto, Japan
179. Toshio Aoki, J.A.E.R.I., Tokai-Mura, Naka-Gun, Ibaraki-Ken, Japan
180. K. A. Mahmoud, Radiation Protection and Civil Defense Department, Atomic Energy Establishment, Abou-Zaabal Post Office, Cairo, U.A.R., Egypt
181. Research and Development Division, AEC, ORO
182-584. Given distribution as shown in TID-4500 under Health and Safety category (100 copies — CFSTI)